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GIF-017-00

Generation IV Roadmap

Description of Candidate Liquid-Metal-Cooled Reactor Systems Report

Issued by the Nuclear Energy Research Advisory Committee and the Generation IV International Forum

December 2002

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ACRONYMS

AFR	advanced fast reactor
ALMR	advanced liquid metal reactor
ATWS	anticipated transient without scram
BREST	Literally, Russian for fast reactor, lead cooled
CDA	core disruptive accident
CRBRP	Clinch River Breeder Reactor Project
DHRS	decay heat removal system
DOE	U.S. Department of Energy
DOE-NE	Department of Energy Office of Nuclear Energy, Science and Technology
DPA	displacements per atom
DRACS	Direct Reactor Auxiliary Cooling System
EBR-II	Experimental Breeder Reactor-II (ANL-W site, US)
EFPD	effective fuel power days
EFR	European fast reactor
EMG	Evaluation Methodology Group
EMP	electromagnetic pump
FCCG	Fuel Cycle Crosscut Group
FFTF	Fast Flux Test Facility (Hanford site, U.S.)
FP	fission product
GEM	gas expansion module
HCDA	hypothetical core disruptive accident
HLLW	high-level liquid waste
HLW	high-level waste
IFR	integral fast reactor
IHTS	intermediate heat transport system
IHX	intermediate heat exchanger
IRACS	Intermediate Reactor Auxiliary Cooling System
LBE	lead-bismuth eutectic (coolant)
LLC	long-life core
LLFP	long-lived fission products
LMFBR	Liquid Metal Fast Breeder Reactor (an archaic term for fast reactors)
LME	liquid metal embrittlement
LMR	liquid metal reactor

LOF	loss of flow
LWR	light water reactor
MA	minor actinide
MABR	Minor Actinide Burner Reactor
MOX	mixed uranium-plutonium oxide (fuel)
NI	Nuclear Island
NOAK	nth of a kind
NPP	nuclear power plant
NSSS	nuclear steam supply system
O&M	Operations and Maintenance
ODS	oxide dispersion-strengthened (cladding)
PAG	planning action guidelines
PHTS	primary heat transport system
PRA	probabilistic risk assessment
PRACS	primary reactor auxiliary cooling system
PRISM	power reactor, inherently safe module
PSAR	preliminary safety analysis report
PWR	pressurized (light) water reactor
RBCB	run beyond cladding breach
RCS	reactor control system
RFI	request for information
RIT	roadmap integration team
RVACS	reactor vessel auxiliary cooling system
SAR	safety analysis report
SASS	self-activated shutdown system
SCNES	self-consistent nuclear energy system
SG	steam generator
SVR	sodium-void reactivity
TRU	transuranic (i.e., Z number >92)
TWG	Technical Working Group
ULLC	ultra-long-life core

EXECUTIVE SUMMARY

This report discusses the first round of R&D roadmap activities of the Generation IV (Gen IV) Technical Working Group (TWG) 3, on liquid metal-cooled reactors. Liquid metal coolants give rise to fast spectrum systems, and thus the reactor systems considered in this TWG are all fast reactors. Gas-cooled fast reactors are considered in the context of TWG 2.

This first round activity is termed "screening for potential", and includes collecting the most complete set of liquid metal reactor/fuel cycle system concepts possible and evaluating the concepts against the Gen IV principles and goals. Those concepts or concept groups that meet the Gen IV principles and which are deemed to have reasonable potential to meet the Gen IV goals are being passed to the next round of evaluation.

Although we sometimes use the terms "reactor" or "reactor system" by themselves, the scope of the investigation by TWG 3 includes not only the reactor systems, but very importantly the closed fuel recycle system inevitably required by fast reactors.

The response to the DOE Request for Information (RFI) on liquid metal reactor/fuel cycle systems from principal investigators, laboratories, corporations, and other institutions, was robust and gratifying. Thirty three liquid metal concept descriptions, from eight different countries, were ultimately received. The variation in the scope, depth, and completeness of the responses created a significant challenge for the group, but the TWG made a very significant effort not to screen out concepts early in the process.

With the number and diversity of concepts submitted in response to the RFI, it was incumbent upon the TWG to seek a grouping of concepts within some logical framework. The rationale for placing concepts into groups was simply that thirty three concepts would be impossible to screen individually with the resources available, and further that this was not needed anyway because some concepts shared many attributes. The issue was how to define the grouping so that a maximum number of concepts fit within a given group, without making the group definition so broad that it would lose practical significance. It was accepted from the start that individual concepts would likely be left over after grouping that would be subjected to individual screening.

The principal guiding criterion for grouping was geared to the product of this specific Gen IV activity: an R&D roadmap. Thus it seemed natural to seek groupings on the basis of common R&D requirements, very loosely defined at this stage. The TWG never intended to use the concept grouping approach as a means to dilute or destroy individual concept attributes. While retention of individual concept attributes may be imperfect or vague in preliminary rounds, it is the TWG intent to retain these individual attributes, and to highlight their individual R&D requirements.

Most of the concepts were initially assigned to one of five concept groups, A through E. In the first round of evaluation the job was to understand the technologies employed, noting specific differences or features that were unique to a specific concept. A cursory evaluation was done of the status of the technologies attendant to the concept group, and finally a preliminary evaluation was done of the R&D that would be required to bring the concept group to deployment reality. This latter subject, the R&D requirements, will be much more extensively addressed in later stages of the roadmap activity. Nevertheless, the TWG felt that at least a limited look at R&D requirements in the first phase would materially aid the process of screening for potential, and that this would provide a useful start to the next phase.

A sixth evaluation, in three parts, was also done. These were termed “base technology evaluations” for fuels, coolants, and fuel cycles. This was done in order to avoid separate subgroups of the TWG doing redundant evaluations as part of the concept group analyses.

With these considerations in mind, five concept groups encompassed 27 of the 33 three concepts submitted:

- Concept group A: medium-to-large sodium-cooled, mixed-oxide fueled reactors with advanced aqueous recycle technology and ceramic pellet or vibratory compaction fabrication (5 concepts).
- Group B: medium-to-large sodium-cooled, metal-fueled (U-TRU-Zr metal) reactors with electrochemical fuel cycle technology (pyroprocessing) (6 distinct concepts)
- Group C: Medium-sized Pb or Pb-Bi cooled; MOX or Th-U-TRU-Zr metal alloy fueled reactors (one concept had nitride fuel); pyroprocess fuel cycle for the metal-fueled concepts, advanced aqueous or unspecified “dry” process for the ceramic-fueled concepts. (9 concepts)
- Group D: Small, Pb or Pb-Bi cooled; metal or nitride fueled reactors with long-life “cartridge” or cassette cores. Fuel cycles vary. (4 concepts).
- Group E: Sodium-cooled concepts that eliminate the traditional secondary sodium loops by development of novel new steam generators. (3 concepts)

In addition, one concept (the Self Consistent Nuclear Energy System, or SCNES) was more a statement of fuel cycle principles. Rather than an evaluation, it was considered in the context of the fuel cycle technology. Five concepts were evaluated by themselves (three direct energy conversion schemes, a concept involving the CANDLE burnup approach, and a concept that would develop Russian Pb-Bi submarine reactor technology for commercialization). Only the Russian submarine technology concept was retained from this set, even then only for a closed fuel cycle version not advertised extensively by the concept sponsors.

Two things seem apparent from the grouped concepts. First, the technology maturity generally decreases from group A to group E. Put another way, group A is nearer-term than group B, etc. Second, there is more similarity in the technical features and in the R&D requirements within groups A and B than in groups C, D, and E.

“Advanced aqueous” recycle technology or the electrochemical pyroprocess were adopted in the vast majority of concepts. Both aim to avoid separation of pure plutonium and to recycle TRU. Both technologies will require considerable development. Use of lead or lead-bismuth coolant has been done successfully in Russia, but the technology is little known in the rest of the world. Corrosion and erosion control and pumping power are concerns, and depending on the specific concept, seismic and other structural issues require resolution. These coolants permit higher temperatures to be reached (one concept that is aimed at production of hydrogen has core outlet temperature of 1050 K) if fuels, cladding, and structural material challenges can be solved. Of course, of the candidate fuels, mixed oxide is well developed (cited as the reference or backup fuel in 10 concepts), with metal fuels requiring continued development (reference or backup in 16 concepts) and nitride fuel requiring essentially complete development (6 concepts).

Not surprisingly, since all of the concepts are fast reactors, all have the capability to utilize almost 100% of the uranium resource, and so a Gen IV goal of resource sustainability is met by all TWG 3 concepts with respect to fuel supply. Uranium mining can be avoided for decades for fast reactor fuel supply simply by using existing stocks of depleted uranium, reducing environmental impact of mining and milling. Enrichment, with it’s environmental, safety, and health impacts, is never needed. In all but a very few cases the high level wastes contain very little plutonium and minor actinides, recycling these material to the reactors. This eases the technical requirements on repositories and reduces the volume of high-level wastes sent to repositories, compared to LWRs operated once through. However, since essentially all the fast reactor concepts capture these advantages, there is little discrimination afforded amongst concepts. Some discrimination is possible wherever real variation exists: size, temperature, modular versus monolithic, specific fuels employed (from something as well-developed as MOX to a more complex Th-U-TRU-Zr metal alloy), and safety. While in general there is a pervasive theme to base the safety case on intrinsic or inherent safety characteristics, the general design features adopted to accomplish this vary considerably.

A number of Gen IV criteria deal with economic potential, and the great challenge to make these systems cost-competitive. The general trends are toward simplification of both reactor systems and fuel cycle technologies; involving smaller space, fewer components, less commodities, less nuclear safety-grade design, etc.

Although a useful start was made in this preliminary round on identification of R&D needed to bring the concept groups to a deployment (or serious development) state, considerably more work needs to be done to put these R&D requirements on a more equal footing, with a more uniform approach.

Description of Candidate Liquid-Metal-Cooled Reactor Systems Report

1. INTRODUCTION

This report documents the first round of R&D roadmap activities of the Generation IV (Gen IV) Technical Working Group (TWG) 3, on liquid metal-cooled reactors. Since liquid metal coolants give rise without exception to high-energy or "fast" spectrum systems, the reactor systems considered in this TWG are all fast reactors. Note, however, that there are other fast reactors under consideration in the Gen IV R&D roadmap, in particular within TWG 2 on gas-cooled reactors, where gas-cooled fast reactor concepts are being evaluated.

This first round activity is termed "screening for potential." It includes (1) collecting the most complete set of liquid metal reactor/fuel cycle system concepts possible, (2) evaluating the concepts against the Gen IV principles and goals, and (3) passing those concepts or concept groups that meet the Gen IV principles and which are deemed to have reasonable potential to meet the Gen IV goals, to the next round of evaluation, the "R&D Scoping" phase.

Although we sometimes use the terms *reactor* or *reactor system* by themselves, the scope of the investigation by TWG 3 includes not only the reactor systems but, very importantly, the entire fuel cycle system as well. Unlike thermal reactors, which can operate on a once-through or partial-recycle basis, fast reactors inevitably require a closed fuel cycle. One cannot evaluate the feasibility of a proposed fast reactor concept without simultaneous evaluation of its complete fuel cycle. For this reason, TWG 3 has developed a special interface with the Fuel Cycle Crosscut Group (FCCG), facilitated by the membership of an FCCG co-chair, Dr. David Wade, on TWG 3. Much more fuel cycle technology discussion will be found within TWG 3 discussion and analysis than is likely within the other TWGs, even at the risk of duplicating material that will be found in FCCG reports.

There have been, thus far, four meetings of TWG 3:

- February 20-21, 2001; Denver, Colorado
- May 8-9, 2001; Lisle, Illinois
- June 13-14, 2001; Las Vegas Nevada
- August 21-22, 2001; Seattle, Washington.

The first meeting was primarily organizational, and preceded the naming of the international members of the group. All but the Las Vegas meeting were joint meetings of all Gen IV TWGs and crosscut groups. All meetings after the first one included substantial participation by the international members. The third meeting in Las Vegas was an exclusive meeting of TWG 3.

The first meeting in Denver also preceded issuance of the Request for Information (RFI) by DOE-NE and the Roadmap Integration Team (RIT). At that meeting the U.S. members of the group decided to begin preparation of system concepts, from the open literature, that originated in countries that were not participating in the Generation IV International Forum, and therefore were not as likely to respond to the RFI. As examples, it was undertaken to prepare concept submissions on the Russian BN-800 and BREST systems. It was further decided to prepare these concept descriptions in "RFI format" (that is, in the format that would be specified in the RFI process), and further to submit them formally to the RFI managers in DOE. Thus, in the end, all the TWG 3 system concepts were included in the DOE document system created for Gen IV. As it turned out, the two Russian concepts mentioned above were the only ones developed by TWG members. The rest were submitted by the principal investigators.

The response to the RFI on liquid metal reactor/fuel cycle systems from principal investigators, laboratories, corporations, and other institutions, was robust and gratifying. Thirty-three liquid metal concept descriptions, from eight countries, were ultimately received. Not surprisingly, there was a great variation in the scope, depth, and completeness of the responses. Some respondents provided supplemental papers and documents, but many did not. Some respondents made clear the intended fuel cycle technologies, and others did not. There were a number of "partial concepts" submitted. Factors such as these created a significant challenge for the group, but the TWG made a very significant effort not to screen out concepts early in the process. Ultimately, only four concepts of the 33 were screened out in this screening-for-potential round.

The remaining concepts were assigned to one of five concept groups, A through E. The groups are described in Section 2. Section 3 summarizes an evaluation of each concept group, the details of which can be found generally in Appendices that are designated by title to match the concept group, i.e., Appendix A is the detailed evaluation of concept group A, and so on. The organization of the evaluation, in both Section 3 and in the Appendices, is to describe the concept group, noting specific differences or features that are unique to a specific concept, to make a preliminary evaluation of the potential of the concept to meet Generation IV goals, to describe the technical uncertainties that are attendant to the concept group (again, noting significant differences between concepts in the group), and finally to provide at least a brief overview of the R&D that would be required to bring the concept group to deployment reality. This latter subject, the R&D requirements, will be much more extensively addressed in later stages of the roadmap activity. Nevertheless, the TWG felt that at least a limited look at R&D requirements in this phase would materially aid the process of screening for potential, and that this would provide a useful start to the next phase, the "R&D Scoping Phase."

In addition to the evaluations of the five concept groups (and one stand-alone concept; see below in Section 2), another evaluation, in three parts, was also done. "Base technology evaluations" were done for fuels, coolants, and fuel cycles. This was undertaken to avoid separate subgroups of the TWG doing redundant evaluations as part of the concept group analyses. These base technology evaluations are summarized in Section 4 and reported in detail in Appendix E.

Section 3 concludes with a "score sheet," for each concept group, of potential against the formal goals adopted by the Gen IV leadership, using the criteria, metrics, and methodologies recommended by the Evaluation Methodology Group. In this round, the metrics are qualitative, and the scoresheets presented in Section 3 reflect results of in-depth discussion amongst TWG 3 members and are thus a consensus view of the professionals involved.

Section 4 summarizes the base technology evaluations, of particular importance in laying out the considerable volume of R&D that crosscuts concepts and concept groups.

Finally, Section 5 discusses the conclusion of this phase of TWG 3 activity.

2. DESCRIPTION OF CONCEPTS AND CONCEPT GROUPS

As noted above, 33 concepts resulted from the response to the RFI, and from unilateral submission by TWG 3. Table 1 summarizes information about the 33 concepts.

With the number and diversity of concepts submitted in response to the RFI, it was incumbent upon the TWG to seek a grouping of concepts within some logical framework. The rationale for placing concepts from Table 1 into groups was simply that 33 concepts would be impossible to screen individually with the resources available, and, further, that this was not needed anyway because some concepts shared many attributes. The issue was how to define the grouping so that a maximum number of concepts fit within a given group, without making the group definition so broad that it would lose practical significance.

It was realized from the start that some concepts might not fit neatly into any useful definition of group boundaries, and so it was accepted from the start that individual concepts might be left over after grouping, which would then be screened individually.

The principal guiding criterion for grouping was geared to the product of this specific Gen IV activity: an R&D roadmap. Thus it seemed natural to seek groupings on the basis of common R&D requirements, loosely defined. It was further realized that such a grouping criterion was not likely going to lead to an unambiguous grouping, and indeed the TWG struggled at times with the issue of the best choice of a group in which to consider a specific concept.

Before moving to the discussion of the specific groups, it is important to note that the TWG has never intended to use the concept grouping approach as a means to dilute or destroy individual concept attributes. While retention of individual concept attributes may be imperfect or vague in this preliminary round, it is the TWG intent to retain these individual attributes into the next round and beyond. To the extent that individual concepts remain included within a group in subsequent rounds of the Gen IV R&D roadmap, it is the intent to highlight their individual R&D requirements.

With these considerations in mind, five concept groups were defined that encompassed 27 of the 33 concepts submitted:

- Concept group A: medium-to-large sodium-cooled, mixed-oxide fueled reactors with advanced aqueous recycle technology and ceramic pellet or vibratory compaction fabrication (5 concepts).
- Group B: medium-to-large sodium-cooled, metal-fueled (U-TRU-Zr metal) reactors with electrochemical fuel cycle technology (pyroprocessing) (6 distinct concepts; with overlap, 8 concepts).
- Group C: Medium-size Pb or Pb-Bi cooled; MOX or Th-U-TRU-Zr metal alloy fueled reactors (one concept had nitride fuel); pyroprocess fuel cycle for the metal-fueled concepts, advanced aqueous or unspecified “dry” process for the ceramic fueled concepts (9 concepts)
- Group D: Small, Pb- or Pb-Bi-cooled; metal or nitride fueled reactors with long-life “cartridge” or cassette cores; fuel cycles vary (4 concepts).
- Group E: Sodium-cooled concepts that eliminate the traditional secondary sodium loops by novel new steam generators (3 concepts).

Table 1. Summary of concepts submitted to TWG 3 on liquid metal reactors.

Concept	Submitted by	Organization	Country	Size (MWe)	Fuel	Coolant	Fuel Cycle	Comment
M1	Boardman	GE	U.S	760	U-TRU-Zr Metal or MOX	Na	Pyroprocess	S-PRISM
M2	n/a	State Scientific Center	Russian Federation	75 – 100	UO2	Pb-Bi	Once-through	SVBR (Submarine reactor)
M3	Sekimoto	Tokyo Institute Center	Japan	`1200 (?)	U-Zr Metal	Na	Once-through (?)	Long-life reactor, CANDLE burnup
M4	Sagayama	JNC	Japan	1500	MOX or Metal (U-TRU-Zr)	Na	Advanced aqueous (or pyroprocess)	JNC sodium cooled fast reactor (JSFR)
M5	Sagayama	JNC	Japan	500	MOX or Metal (U-TRU-Zr)	Na	Advanced aqueous (or pyroprocess)	Modular JNC sodium-cooled fast reactor (M-JSFR)
M6	Lee	University of Michigan	U.S.	800	UO2, MOX (or metal?)	Na	n/a*	BN-800
M7	Hahn	KAERI	Korea	150	U-TRU-Zr Metal	Na	Pyroprocess	KALIMER
M8	Hahn	KAERI	Korea	300-500	U-Pu-Zr Metal	Na	Dry fabrication	Ultra long-life core achieved with recladding
M9	Kim	Pohang University	Korea	n/a	n/a	Liquid Metal	n/a	Liquid Metal MHD
M10	Chang	Korea Adv. Inst. Of S&T	Korea	500	n/a	Na	n/a	Main innovation is steam gen. Integrated with IHX
M11	Greenspan	University of California	U.S.	125 MWt	U-Zr, U-TRU-Zr or Nitride	Pb-Bi	AIROX or Pyroprocess	ENHS. 20-year cartridge core
M12	Paramonov	Westinghouse	U.S.	26	Unknown (high temp)	Na, K, or Li	n/a	Direct conversion (AMTEC)
M13	Sienicki	ANL	U.S.	120-160	U-TRU Nitride	Pb-Bi	Pyroprocess	STAR-LM (liquid metal)
M14	Tsiklauri	PNNL	U.S.	300 (?)	MOX	Pb	Once-through burner	Oxide-fueled version of BREST
M15	Hill	ANL	U.S.	300	U-TRU-Zr Metal	Na	Pyroprocess	AFR-300
M16	Lineberry	ANL	U.S.	300, 1200	U-TRU Nitride	Pb	Dry process or advanced aqueous	BREST
M17	Wade	ANL	U.S.	400 MWt	U-TRU Nitride	Pb	Pyroprocess	STAR-HZ, long-life contridge core
M18	Buongiorno	INEEL	U.S.	419	Th-U-Pu-MA-Zr Metal	Pb-Bi	Pyroprocess	Actinide burner, direct contact steam generator
M19	MacDonald	INEEL	U.S.	~400	Th-U-Pu-MA-Zr Metal	Pb-Bi	Pyroprocess	Shares numerous attributes of M18

Table 1. (continued)

Concept	Submitted by	Organization	Country	Size (MWe)	Fuel	Coolant	Fuel Cycle	Comment
M20	Matsui	Inst. Applied Energy	Japan	n/a	Metal, MOX, or Nitride	n/a	n/a	SCNES principles
M21	Nascimento	IEAV/IPEN	Brazil	~350	U-TRU metal or nitride	Pb	Pyroprocess	Integrated Lead Reactor
M22	Caronnier	CEA	France	1500	MOX	Na	n/a (presume adv. Aqueous)	RNR-1500
M23	Todreas	MIT	U.S.	~400	Th-U-Pu-MA-Zr metal	Pb-Bi	Pyroprocess	Minor Actinide Burner Reactor
M24	Minato	CRIEPI	Japan	50	U-Zr Metal	Na	Pyroprocess option (?)	4S Long-life core U-TRU-Zr recycle option
M25	Walter	LLNL	U.S.	780 (e.g)	U-TRU-Zr metal	Na	Pyroprocess	Endorsement of M1, M15
M26	Wider	Joint Research Center of EC	Italy	n/a	n/a	Pb or Pb-Bi	n/a	Focus on safe way to put SG in primary vessel
M27	MacDonald	INEEL	U.S.	~400	Th-U-Pu-MA-Zr metal	Pb-Bi	Non-aqueous process	Pebble bed actinide burner
M28	Paramonov	Westinghouse	U.S.	n/a	Unknown (high temp)	K	n/a	Potassium vapor cycle
M29	Alekseev	Kurchatov Institute	Russian Federation	340	MOX	Pb-Mi	Non-aqueous process	RBEC; part of synergestic fast-thermal reactor mix
M30	Lennox	NNC Ltd.	UK	1500	MOX	n/a	n/a	Compact Pool Fast Reactor
M31	Lennox	NNC Ltd.	UK	n/a	n/a	n/a	n/a	Simplified Fast Reactor; copper-bonded SG
M32	Lennox	NNC Ltd.	UK	50MWt	n/a	n/a	n/a	Small Integrated Power Source; In-Vessel SG ,40 year core
M33	Arie	Toshiba	Japan	~600	n/a	n/a	n/a	A metal-fueled fast reactor core for SCNES

* n/a indicates not available, or not specified. See “List of Acronyms” for definition of other terms.

In addition, one concept (the Self Consistent Nuclear Energy System, or SCNES) was more a statement of fuel cycle principles. Rather than an evaluation, it was considered in the context of the fuel cycle technology. Five concepts were evaluated by themselves (three direct energy conversion schemes, a concept involving the CANDLE burnup approach, and a concept that would develop Russian Pb-Bi submarine reactor technology for commercialization). Four of these were rejected from further consideration in the Generation IV roadmap: concepts M9, M12, and M28. M9 (liquid Metal MHD) was eliminated without dissent by TWG 3 because it was viewed as very unlikely to meet the deployment timing schedule of the Gen IV program. M12 and M28 were eliminated, again without dissent, because of nuclear safety deficiencies. The CANDLE concept (M3) was eliminated because it is not sufficiently developed to be considered a system concept. After internal debate, the Russian submarine concept was passed into the next round, and thus it was the only concept treated outside the concept groups as stand-alone in this round.

Two things are apparent from the grouped concepts. First, the technology maturity decreases in passing from group A to group E. Put another way, group A is nearer-term than group B, which is nearer-term than group C, etc. Second, there is more similarity in the technical features and in the R&D requirements within groups A and B than in groups C, D, and E.

Before going on to the summary descriptions and evaluations of the concept groups in the next section, a few observations may provide some perspective. As is seen in Table 1, "Advanced aqueous" processing or the electrochemical pyroprocess were adopted in the vast majority of concepts. Both aim to avoid separation of pure plutonium and to recycle transuranics. Both technologies will require considerable development. Use of lead or lead-bismuth coolant has been done successfully in Russia, but the technology is little known in the rest of the world. Corrosion control and pumping power are concerns, and depending on the specific concept, seismic and other structural issues require resolution. These coolants permit higher temperatures to be reached (one concept that is aimed at production of hydrogen has a core outlet temperature of 1050 K) if fuels, cladding, and structural material challenges can be solved. Of course, of the candidate fuels, mixed oxide is well developed (cited as the reference or backup fuel in 10 concepts), with metal fuels requiring continued development (reference or backup in 16 concepts) and nitride fuel essentially starting from scratch (6 concepts). Issues such as these will be highlighted subsequently.

3. SUMMARY EVALUATION OF THE CONCEPT GROUPS AND STAND-ALONE CONCEPT

This section describes each concept group, including brief discussion of the attributes of individual concepts within the concept group. On a preliminary basis, the following items were generally evaluated (by subgroups of members of the TWG), as encouraged by the RIT:

- Potential of the concept to meet the Generation IV goals
- Technical uncertainties
- Overall concept potential versus R&D risk.

This was done for each of concept groups, A through E, and is summarized in Section 3.1 (Subsections 3.1.1 through 3.1.5). As noted above, the more detailed evaluations are presented in Appendices A through D (due to lack of specific design information, no detailed Appendix is provided for concept group E). For concept M2, the Russian SVBR concept based on submarine technology, the evaluation is found in Subsection 3.1.6, and no detailed Appendix was prepared for this concept either.

Then, presented in Section 3.2 are the screening-for-potential "scoresheet" evaluations, in the format developed by the EMG, for the four major concept groups (A through D).

3.1 Concept Group A

Concepts in Group A are medium-to-large sodium cooled fast reactors, with mixed uranium-plutonium dioxide (MOX) fuel. The sizes vary from 500 to 1500 MWe. The concepts included in group A are depicted below in Table 2. Refer to Table I in Section 2 for more detail, or better, to Appendix A.

Table 2. Group A concepts.

Number	Concept Name	Sponsorship
M4	JNC Sodium-Cooled Fast Reactor (JFSR)	JNC
M5	Modular JNC Sodium-Cooled Fast Reactor (M-JFSR)	JNC
M6	BN-800	University of Michigan*
M22	RNR 1500	CEA
M30	Compact Pool Fast Reactor (CPFR)	NNC, Ltd.

* Submission prepared by Prof. John Lee, University of Michigan.

The fuel cycle intended is generally "advanced aqueous" recycle technology, with an advanced form of pellet fabrication or vibratory compaction of fuel particles. Advanced aqueous recycle can mean different approaches in different countries. The fuel cycle for the Russian concept (BN-800) may in fact be their oxychloride electrochemical process coupled with vibratory compaction. The fuel cycle intended for the CPFR (M30) was unspecified.

The group A concepts represent, more or less, the traditional line of fast reactor development (MOX fuel, aqueous recycle) followed for more than three decades in the national programs in France, the United Kingdom, Japan, and Russia. This is not to say that considerable new work has not been done. Quite the opposite is true: the aqueous recycle technology is not the traditional PUREX, and significant work has gone into simplifying the plant design in order to reduce cost. Nevertheless, the basic characteristics of the systems in this group would qualify them as nearer-term Gen IV concepts than the other concept groups in TWG 3.

There are a number of similarities amongst the five concepts, in addition to the MOX fuel:

- Core outlet sodium temperature ~820K
- Thermal efficiency about 40%
- Core height ~1m

Also, advanced safety features are common to all the concepts in the group

- Diversity in the reactor control system, utilizing passive modes of insertion
- Diversity and redundancy in the decay heat removal system, utilizing natural circulation.

The major R&D needs of the systems in concept group A relate mainly to the development of the case for passive safety, and, overall, to the development of cost-reduction features. In addition, the fuel cycle technology has many issues outstanding. Nevertheless, group A concepts entail the least R&D risk of any of the TWG 3 concept groups, and should certainly be accepted for more detailed analysis and evaluation.

3.2 Concept Group B

Concepts in group B are medium-size (modular) sodium-cooled fast reactors fueled with U-TRU-Zr metal alloy fuel and utilizing the pyroprocess fuel cycle. An exception is concept M4, which is a large monolithic reactor with an output (1500 MWe) well beyond the general 300–500 MWe range of this concept group. Concepts M4 and M5 were included in both concept group A (oxide fuels, advanced aqueous recycle technology) and concept group B (metal fuels, pyroprocess) simply because metal fuel is addressed as an advanced option for enhancement of core performance in the Japanese program.

The concepts included in group B are depicted below in Table 3. Refer to Table 1 in Section 2 for more detail, or better, to Appendix B.

Table 3. Group B concepts.

Number	Concept Name	Sponsorship
M1, M25	S-PRISM	GE, LLNL
M4	JNC Sodium-Cooled Fast Reactor (JSFR)	JNC
M5	Modular JNC Sodium-Cooled Fast Reactor (M-JSFR)	JNC
M7	KALIMER	KAERI
M8	ULLC achieved with recladding	KAERI
M15, M25	Advanced Fast Reactor, AFR-300	ANL, LLNL
M33	Metal-fueled fast reactor for SCNES	Toshiba

The group B concepts therefore generally represent the technology directions advocated by Argonne National Laboratory in its 1984–94 Integral Fast Reactor (IFR) program, although it was by no means exclusively aimed at medium-sized modular concepts.

The reactor technologies represented in group B are firmly rooted in the successful experience in the United States with EBR-II and, less directly, with FFTF. The key issue, from the reactor perspective, is the concept of modularity. Can the economies of scale that benefit the large plants be overcome with economies of modular systems such as factory fabrication? For the pyroprocess fuel cycle, its economic promise due to its simplicity and compactness is clear, but critical development work (e.g., U-Pu co-extraction experiments) was cut off in 1994, even though much of the core technology development continued as a spent fuel treatment-for-disposal technology.

Group B concepts perhaps entail more risk of development then group A, but much less risk than the concepts included in groups C, D, and E. Moreover, the potential for cost reduction, further improvement in safety, the easing of repository technical requirements, and improved proliferation resistance, easily qualify the group B concepts for passage to the next round.

3.3 Concept Group C

Nine concepts were considered within this group. They each share the common characteristics of Pb or Pb-Bi coolant, and medium size (all are in the range 300–400 MWe). Refer to Table 1 in the Introduction section, and to Appendix C. Table 4 presents the concepts included in Group C.

Table 4. Group C concepts.

Number	Concept Name	Sponsorship
M14	Oxide-fueled version of BREST	PNNL
M16	BREST	MINATOM
M18	Actinide burner; direct contact steam generators	INEEL
M19	LWR Spent Fuel Actinide Burner	INEEL
M21	Integrated lead reactor	INEEL
M23	Minor Actinide Burner Reactor	MIT
M26	Safe way to include SG in primary vessel	Res. Center EC
M27	Pebble bed actinide burner	INEEL
M29	RBEC	Kurchatov Institute

Of the nine concepts in the group, four are related and were submitted by the same INEEL/MIT group. Counting these submittals as a single group for the moment, and discounting M26 (which deals only with a component issue), we have four distinct concept types in this grouping. Half use Pb-Bi, and half Pb. MOX is the fuel in two cases, metal in one (the four similar concepts), and nitride in the remaining concept.

Three major technical areas (common to all lead-alloy reactor concepts) are identified that are in need of extensive research and development before these reactors can be deployed. These areas are (1) neutronics core design, (2) fuel performance and (3) compatibility of the structural materials with the coolant. In addition, several of the proposed concepts adopted actinide burning as a performance goal, which is different from conventional fast reactor consideration. In somewhat more detail, the issues are:

A major emphasis will be placed on core controllability because of the small delayed neutron yield and Doppler reactivity feedback of the cores with high minor actinide loadings. Calculations showed that these important safety parameters are compromised in a fertile-free core; hence, the addition of fertile material is necessary.

For the reactor concepts utilizing MOX fuels, there is a need to study the compatibility and interaction of the fuel with Pb or Pb-Bi. For the reactor concepts utilizing advanced fuel forms that appear most promising (nitride and thorium-based metallic fuels) and offer additional performance benefits, little knowledge exists or significant extrapolation of existing technology is needed, both requiring significant R&D. It is recognized that a need exists for better knowledge and understanding of the basic properties of the fuels prior to and during irradiation (e.g., phase diagrams, thermal conductivity, diffusion coefficients, swelling characteristics, fission gas release rates, restructuring characteristics, etc.).

The use of heavy metal coolants in a power-producing reactor strongly depends on the corrosion resistance of the structural materials, in particular the fuel cladding. If an accelerated deployment schedule is to be pursued for any heavy-liquid-metal reactor concept, the work in the cladding and structural materials area should be expanded/organized to identify suitable materials for the fuel, fuel cladding and the core internals. Also, operating envelopes for these materials need to be generated as a function of coolant type, temperature, fast fluence, burnup and oxygen concentration.

The direct contact of lead-bismuth and steam significantly aggravates the issue of coolant activation. The primary and secondary coolants (lead-bismuth and water, respectively) are not physically segregated and a substantial amount of radioactive polonium (the main product of bismuth neutron activation) may be released into the secondary system. This may make access and maintenance of the power cycle components costly. The concentration of polonium in the primary coolant (and thus the release of polonium to the steam) can be reduced significantly by making use of an online polonium extraction system. Some potentially effective polonium extraction techniques have been identified, but they need extensive R&D.

The pyroprocess is stated as the fuel cycle technology of choice for the several concepts employing Th-U-Pu-MA-Zr metal fuel, yet no electrochemical process flow sheet has been proposed for such a fuel choice.

Concept M18 utilizes a direct contact steam generator that is significantly different from the conventional steam cycles and has the potential for significant simplification and cost reduction. The separation of Pb-Bi and steam in the steam dryer of Concept M18 is not complete. A small amount of Pb-Bi aerosol remains entrained in the steam stream and is carried over to the turbine, which may cause liquid metal embrittlement of the stressed parts of the turbine (e.g., the blades and casing).

3.4 Concept Group D

The concepts in group D (listed below) are all simple, modular systems centrally fabricated and fueled with long-life cores (refer to Table 1 in Section 2, or to Appendix D), transported on site, operated with minimum local nuclear infrastructure and requirement of expertise, and retrieved for refueling and waste management. Concept M17 goes beyond electricity generation to devise applications in hydrogen generation, and utilization of waste heat for potable water production.

One common implicit (but not exclusive) underlying assumption may be that these systems might be deployable in the intermediate term, in countries with minimum nuclear power infrastructure. Another common philosophy may be that the smaller modules will better meet the incremental market needs with lowered financial risks. The cartridge or cassette-type refueling with regionalized fuel cycle services demands major technical and institutional considerations.

Table 5. Group D concepts.

Number	Concept Name	Sponsorship
M11	Encapsulated Nuclear Heat Source (ENHS)	UCB & LLNL
M13	Secure Transportable Autonomous Reactor (STAR-LM)	ANL
M17	Secure Transportable Autonomous Reactor (STAR-H2)	ANL
M24	Super Safe, Small, Simple LM Reactor (4S)	CRIEPI

The common design features of group D are small, modular-size pool type reactors with:

- Lower power output—ranging from 125 to 400 MWth
- Low power density
- Long refueling intervals—15 to 30 years
- Lead or lead/bismuth coolants (except M24: sodium)
- Metallic or nitride fuel—U, Pu, MA, LLFP, no blanket
- Fuel cartridge/cassette factory-fabricated and overland-transportable including reactor internals, intermediate heat exchanger and electromagnetic pump for M24
- No on-site fuel cycle facilities (regionalized fuel cycle services)
- Natural convection flow for primary heat transport (except M24)
- Autonomous following of generator load variations for a wide range of nominal power
- Elimination by design of severe accidents scenarios leading to core damage
- Passive decay heat removal and passive containment vessel cooling
- Simple design (no pumps and intermediate heat exchanger; except M24)
- No on-site refueling mechanism
- No mechanical connection among the fuel cartridge/module and the steam generators
- Simple reactivity control systems (movable reflector mechanism)
- No safety function for the balance of plant.

The R&D needs for group D are, like group C, dominated by issues arising with Pb or Pb-Bi coolant (see Section 4.2, and Section E.2 of Appendix E). Major issues include:

- Corrosiveness and erosiveness of Pb-Bi, and control techniques
- New materials R&D, not only because of the corrosive properties of the coolant, but also because of the higher temperatures generally associated with Pb coolant.
- Handling of the Po 210 produced in Pb-Bi coolant
- Demonstration of natural circulation cooling capability
- The reflector drive mechanism for reactor control
- The fuel cartridge approach to refueling
- Regionalized fuel cycle centers
- Fuel cycle technology with nitride fuel.

3.5 Concept Group E

This group deals not with complete system concepts, but rather with three concepts for eliminating the intermediate (sodium) heat transport system (IHTS) in sodium-cooled reactors. This IHTS has been included in all sodium-cooled power reactors to mitigate the consequences of steam generator leaks and resultant sodium-water chemical reaction. If such were to occur with the IHTS present, the reaction products would involve only natural sodium. Absent the IHTS, the reaction products would include the Na-24 (half-life = 15 hr) present in the primary system as an activation product. The motivation for IHTS elimination is cost reduction.

Three concepts are included in this group, shown below in Table 6 (refer to Table 1 in the Introduction section for more detail):

Table 6. Group E concepts.

Number	Concept Name	Sponsorship
M10	Ultra Long-Life Sodium-Cooled Reactors with Steam Generators Integrated	KAIST
M31	Simplified Fast Reactor	NNC Ltd.
M32	Small Integrated Power Source	NNC Ltd.

It is not clear to what group these concepts might be assigned were it not for their sharing the attribute of IHTS elimination in a sodium-cooled system. None of the three concepts included information about the fuel or fuel cycle.

4. CONCEPT DESCRIPTIONS AND COMPARISONS

Concept M10 integrates the steam generator within what is termed an intermediate heat exchanger, which is located in the primary vessel. Many rather complex tubes (i.e., three embedded tubes) provide the heat exchange. Reactor outlet sodium is circulated down the periphery of each tube by an EM pump. Each tube has stagnant inner layer of Pb-Bi, with another dual tube inside with a water downcomer being the most interior tube, and a water/steam riser tube outside. Clearly, the Pb-Bi layer is intended as a buffer between the water and the sodium.

Concept M31, the Simplified Liquid Metal Fast Reactor (SFR), uses a copper bonded heat exchanger to eliminate the IHTS in what appears to be a large-sized loop-type fast reactor. The heat exchanger is a series of water and sodium tubes, all embedded in metallic copper. Three barriers are thus claimed which prevent the sodium-water reaction: the water tubes, the sodium tubes, and copper metal filling continuously the interstices between both kinds of tubes.

Concept M32, by the same sponsors as M31, is the Small Integrated Power Source (SIPS). The system as proposed is 50MWt and has a nominal 40-year core life. It uses the same basic copper-bonded heat exchanger concept as M31, only here located within the primary vessel. It is relatable to the 4S concept (M24) in concept group D, but the elimination of the IHTS in this concept is why it is located in group E.

5. CONCEPT POTENTIAL AND TECHNICAL ISSUES

In terms of potential to meet the Gen IV goals, the differentiating features of Group E are in safety and economics. One would accept cost reduction potential because of IHTS elimination, but these heat exchangers are complex and their fabricability requires further study.

The TWG is aware of a number of prior attempts, in the United States at least, to invent concepts for elimination of the IHTS. Westinghouse was reported to have once had a government-funded program in this area. Apparently, the results of this formal effort (and informal ones reported to the TWG by its General Electric Co. members) were insufficient to warrant pursuit at that time.

The key feasibility issue with elimination of the IHTS is safety. The consequences of a steam/water reaction would be greater, if radioactive primary sodium were involved. To be risk-neutral, seemingly a minimum requirement, the probability of such reactions must be reduced, perhaps greatly reduced, from those in conventional sodium-cooled fast reactor steam generators.

Nevertheless, in this screening-for-potential round, the TWG opts to retain concept group E for the next round.

5.1 Concept M2: Small Fast Lead-Bismuth Cooled Reactor

SVBR-75/100 is a small power reactor module cooled by lead-bismuth eutectic (LBE) that can, according to its sponsors, be demonstrated and deployed within the next decade. It has generated considerable debate within the TWG, principally because of its aim for next-decade deployment, and therefore as to whether it satisfies the Gen IV principles as a truly advanced reactor system. Moreover, the reference fuel and fuel cycle for this system, according to its sponsors, is enriched uranium dioxide operating on a once-through cycle. If deployed in this fashion, it would consume uranium resources at roughly three times the rate of an LWR operated once through. However, the sponsors make brief mention of a MOX-fueled option for this system, which is conceded to be a likely feasible option. The TWG must then conjecture about its fuel cycle technology, for none is specified. The uranium-fueled once-through option was rejected out-of-hand by the TWG, but the ill-defined plutonium fueled option was retained.

This reactor design is based on the Russian LBE nuclear coolant technology successfully developed and deployed in the past several decades. Eight alpha-class nuclear submarines have utilized this reactor technology. The concept is a two-circuit reactor of integral design. Reactor vessel dimensions are roughly 4.5 m in diameter and about 7 m in height. All primary circuit components are housed within the reactor vessel. Mechanical pumps provide LBE circulation. Modular steam generators with bayonet-type tubes are integrated into the reactor vessel. It can produce saturated or superheated steam at temperatures up to 400 C. Reactor power is 100 MWe or less, depending on restrictions placed by rail shipability of the reactor vessel.

As for potential for meeting Gen IV goals, this concept would seem to be viable as a Gen IV system only with a plutonium-based fuel. This is a cartridge-like core, with the core, control rods, and coolant transported and replaced together; in that sense it shares attributes of concept group D.

Total core void reactivity is negative (in the uranium-fueled mode), and the maximum local positive void reactivity is less than $1. Excellent negative feedback effects are expected in all accident scenarios. In case of accidents with failure of all secondary system capabilities, passive heat removal is attained through the heat removal system installed in the water storage tank around the reactor vessel.

Because an LBE cooled reactor typically has a large heat capacity, the thermal response time constant is considered to be relatively high. However, the coolant volume of this concept is very small (190 tons coolant/100 MWe), and it is therefore necessary to confirm the plant safety characteristics.

The reported capital costs vary among different reports but are generally low. An earlier nuclear island replacement proposal reported a $560 per kWe installed capacity (not including the balance of the plant).

The TWG judges that there is value to carry this concept along, because its basic R&D needs are essentially the same as for all of the other Pb and Pb-Bi systems included in the concept groups C and D. Additionally this reactor design (not fully represented in the submitted proposal) is the most mature concept that is associated with Pb-Bi coolant technology.

5.2 Scoresheet Evaluation of the Concept Groups

This section collects the scoresheets developed by consensus within the TWG. The purpose of the narrative in this section is to provide at least a basic rationale for the choices made by the TWG.

First, with regard to fuel utilization, all of the TWG 3 concepts are fast reactors, and therefore nearly all have the inherent capability to utilize fully the uranium resource, less only heavy metal losses to waste during the infrequent recycle. While breeding ratios and doubling times vary amongst the concepts, to first order this affects only the growth rate that is possible. Thus the Gen IV goal of sustainability is met with respect to fuel supply by all TWG 3 concepts. All concept groups were given a top score for criterion SU1-1 for fuel utilization on the scoresheets.

SU1-2 measures fuel cycle impact on the environment. The closed fuel cycles of the TWG 3 concepts involve some incremental environmental impact over the ALWRs operated once-through. The TWG contends, however, that the environmental impacts associated with mining, milling, enrichment and storage of depleted uranium, all foregone with fast reactors, exceeds the environmental impacts of the closed fuel cycle. All four concept groups also received top scores for this metric.

The next criterion, SU1-3, looks at “utilization of other resources.” For the sodium-cooled systems, the commodities are concrete, steel, and sodium. None are in short supply, similar to the situation for commodities for ALWRs, and thus groups A and B are ranked “similar to reference”. Lead-bismuth systems put demands on the relatively scarce bismuth resource, although in the Japanese fast reactor feasibility assessment bismuth scarcity is not thought to be that large an issue. Nevertheless, the TWG rated groups C and D slightly worse than the reference in this regard.

Sustainability goal 2 involves criteria for waste minimization (SU2-1), environmental impact (SU2-2), and stewardship burden (SU2-3). The closed fuel cycles of TWG 3 all imply much less HLW to be disposed than LWRs operated once through. All concept groups were given a top score in this regard, although the uncertainty was extended downward in the group C and D cases because of the uncertainty associated with polonium and possibly C-14. In terms of environmental impact, for sodium-cooled systems the TWG felt that there is insufficient information to make a meaningful comparison between the concept groups and the reference, so a neutral rating was assigned. While these same uncertainties prevail for lead-cooled systems too, it was felt that there is some potential for negative impact because of the toxicity of lead, and the generation of polonium-210; and the bands in the scoresheets reflect this.

With respect to stewardship burden (SU2-3), the closed fuel cycles under consideration by TWG 3 all involve recycle of not only plutonium, but minor actinides as well. If fully achieved, this significantly eases the technical requirements imposed on repositories. However, recovery of plutonium and minor actinides will never be 100%. The TWG therefore gave all concept groups a score that extended from somewhat better than the reference to much better. Fuel cycle technologies with potential to yield lower process losses would move to the right on the scoresheet. Also, in a number of countries and especially in Japan, there is interest in and investigation of separation of long-lived fission products from the HLW to be disposed of. If this were to be achieved, it too would move the stewardship burden rating to the right.

Sustainability goal 3 examines the proliferation resistance of the concept groups through criteria for material life-cycle vulnerability (SU3-1), application of extrinsic barriers (SU3-2), and unique characteristics (SU3-3). On vulnerability, the spent fuel is unattractive in both the fast reactor system and in the LWR once-through system. However, most of the fissile material in the fast reactor system is largely in the core of the reactors, which is thought to be a major lessening of vulnerability from that when plutonium is in spent fuel that is either in storage or in repositories distributed all over the world. The scoresheet thus shows a rating of better-than-reference for life-cycle vulnerability. There are, in addition, beneficial unique characteristics (SU3-3) that all of the fuel cycles under consideration by TWG 3 share. For example they are designed so that plutonium is never to be separated in pure form. Minor actinides are to accompany the uranium-plutonium product. In the case of the pyroprocess (group B and certain of the group C and D concepts), the fuel cycle facilities have a potential to be co-located with the reactors, eliminating most transportation. For these reasons, each of the concept groups is rated as better than the reference for unique characteristics. Criterion SU3-2, on application of extrinsic barriers, is problematic. EMG documentation notes that this criterion should not be evaluated at this juncture.

Safety and reliability goal 1 contains criteria related to routine exposures to radiation, chemicals, or toxic hazards (SR1-1); the potential for occupational accidents (SR1-2); and reliability (SR1-3). With regard to radiation exposure, the TWG believes that exposures for reactor plant personnel are likely to be moderately lower in fast reactors than in LWRs. This is thought to be supported by the sparse data available, but more importantly it should be true because of the lesser amount of activated corrosion products (at least in sodium-cooled systems) and greater degree of confinement of radionuclides. The TWG was informed by members of the Fuel Cycle Crosscut Group that exposures of a standard PUREX-type fuel cycle were roughly equal to that of mining, milling, enrichment and spent fuel storage in once-through systems. The advanced fuel cycle technologies of the Gen IV era will improve upon PUREX performance, just as experience with mature LWRs has improved upon early LWR experience. Thus on radiation exposures overall, the TWG believes it is plausible that the fast reactors can enjoy a modest advantage. While sodium is a chemical hazard, experience with it has been very good from the view of hazard to workers. Offsetting the occupational hazards of sodium, perhaps, is the prevalence of high pressure water systems in LWR plants. For these reasons, the TWG rates groups A and B as “better than reference” for SR1-1. With what is now known, the lead cooled systems are likely to be worse than the reference because of the radiation hazard of polonium, and because of the toxicity of lead, and the SR1-1 ratings for groups C and D reflect this assertion.

The TWG could think of no reason why occupational safety in the sodium-cooled systems should be significantly different than the reference, while the lead-cooled systems may be slightly worse because of the same hazard potential as cited above.

Reliability (SR1-3) proved a very tough criterion for the TWG to evaluate. The simple fact is that, over the last decade or two, the capacity factor of the deployed fleet of LWRs has slowly but steadily risen to about the 90% level. It is hard to imagine that any reactor concept would claim that it could do better than this level, at least until the underlying technology had reached a level of maturity roughly equal to that of today's LWRs. For this reason, the TWG simply opted to put the rating on reliability at the level indicating that it could be similar to that of the deployed LWR fleet, but notes parenthetically that it may take decades to get there.

Safety and reliability goal 2 concerns robustness of engineered safety features (SR2-1), the uncertainties associated with system safety modeling (SR2-2), and any unique characteristics to achieve low probability of core damage (SR2-3).

The engineered safety features of the sodium cooled plants, in general, appear very robust and rate a top score, in the TWG's opinion. This results from such things as passive/active reactor shutdown systems and decay heat removal systems, the low-pressure primary system boundary, and the redundancy and diversity of the control system. For example, introduction of Self-Actuated Shutdown Systems (SASS) or Gas Expansion Modules (GEM) into one of two lines of the reactor shutdown system is being discussed. By means of those devices, the frequency of Hypothetical Core Disruptive Accidents (HCDA) is expected be less than 10^{-7} per reactor year. In addition, a decay heat removal system that relies on natural circulation is generally incorporated. Similarly, the system modeling uncertainties are relatively low, given the advanced state of safety research in sodium-cooled systems. For these and other reasons, the probability of core damage is extraordinarily low, and it appears quite possible to design sodium-cooled plants to survive all of the classical ATWS events without core damage.

The Pb and Pb-Bi systems may be able to replicate this safety behavior, but that is far less established and the uncertainty is higher. The scoresheets reflect this.

The TWG evaluated only the first two of the four criteria in safety and reliability goal 3; highly robust mitigation features (SR3-1), and damage/transport/dose understood (SR3-2). With regard to the first criterion, it was recognized that for sodium cooled systems items such as the recriticality-free concept being pursued in Japan for oxide fueled systems, or the additional safety testing thought necessary in the U.S. for metal-fueled systems, could result in a well established and highly robust set of mitigation features. Accordingly, this was rated “better than reference” in both concept groups A and B. Conversely, the TWG felt that the damage/transport/dose mechanisms were roughly equally well known in the sodium-cooled systems as in the ALWRs, so the rating on that criterion was at par. Once again, for the Pb or Pb-Bi systems and their mitigation features, the general feeling of the TWG was that these systems may match the sodium systems with adequate maturation—and the rating was slightly less optimistic.

Finally, there are the economic criteria, of which there are five: capital costs, financial costs, production costs, development costs, and “profitability.”

For sodium cooled reactors extensive efforts has been devoted to cost reduction based on the experiences to date in design and construction. As a result, the reactor design with reduced amounts of commodities has been attained in both large monolithic and medium-sized modular plants. JNC has documented this in the Japanese fast reactor feasibility assessment, and in the U.S., the G.E. PRISM effort resulted in the same conclusion. Thus the capital cost criterion, EC-1, was evaluated as better than reference.

Next, with regard to the financial cost (EC-2), the TWG first assumed that financial costs were related to technology risk, but we were informed by a representative of the EMG that this was simply interest during construction. The construction period is thus of primary interest. The duration of construction of the large monolithic plants was estimated to be comparable or shorter than that of the ABWR (JNC), and that of medium-size modular plant was much shorter (GE and JNC). Because of these facts, the financial cost was evaluated to be better than reference.

As for production cost (fuel cost and O/M cost), existing LWR production costs are very low. However, it was suggested from the analyses in the FCCG that fuel costs of fast reactors are comparable or cheaper than that of LWR. The high end of the score bar reflects the assumed increased uranium prices in 30–40 years from now.

Lead-bismuth systems cannot claim the technological maturity of the sodium systems. Because of this, the score of group C and D are at par or worse than reference, and the wide ranges of the bar reflect the large uncertainties associated with lead-bismuth systems.

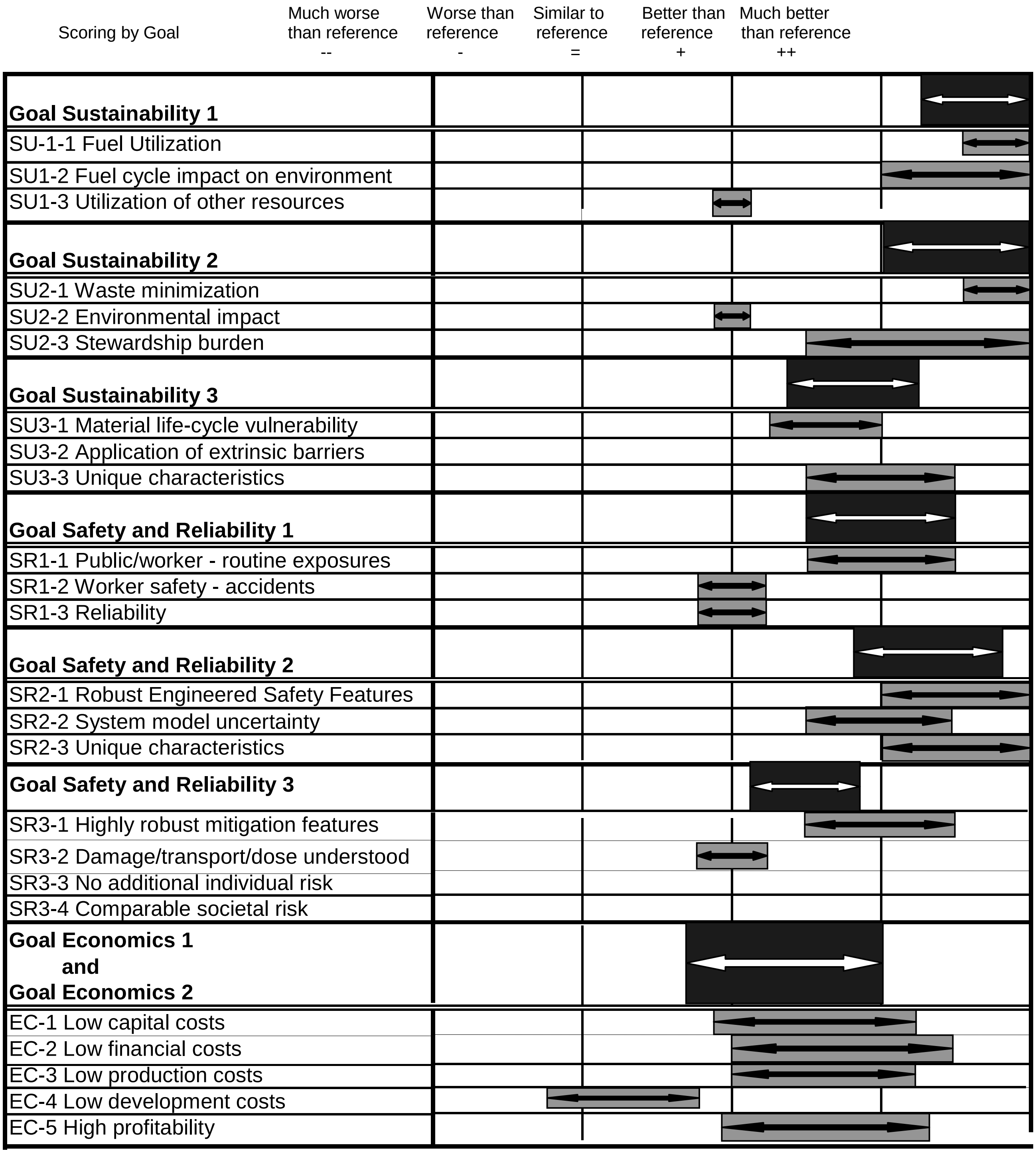
A representative from EMG revealed the scales proposed for the evaluation of development costs. We were told that the value of the reference was \$0.5 billion (the reference value was changed to \$0.75 billion at the final plenary meeting.). According to the proposal by the EMG, all development costs of TWG 3 systems are likely to be to the left of the reference on the scoresheets. A distinction between sodium cooled and Pb/Bi cooled reactors should be kept. The development cost ratings implicitly exclude the development of prototypes and demonstration reactors. TWG 3 believes that it is reasonable and proper to display dollar cost estimates of development, but that this should be balanced by a dollar benefit criterion/metric for the proposed development.

With regard to EC-5 "High Profitability," it was cleared up at the final plenary meeting that the criteria EC-5 meant merely bus-bar costs. Afterwards TWG 3 made an estimate of what the bus bar cost range ought to be based on the estimates of capital cost, interest during construction, and production cost (i.e., fuel and O&M). Then, EC-5 was evaluated merely as a rough average over three criteria, EC-1, EC-2 and EC-3. A problem here is that the item considering the "other profits" such as hydrogen production that is pointed out in the metrics adopted by the EMG, would be eliminated.

Screening for Potential Scoresheet

Concept name: Medium-to-Large Oxide-Fueled Sodium-Cooled Systems (Group A)

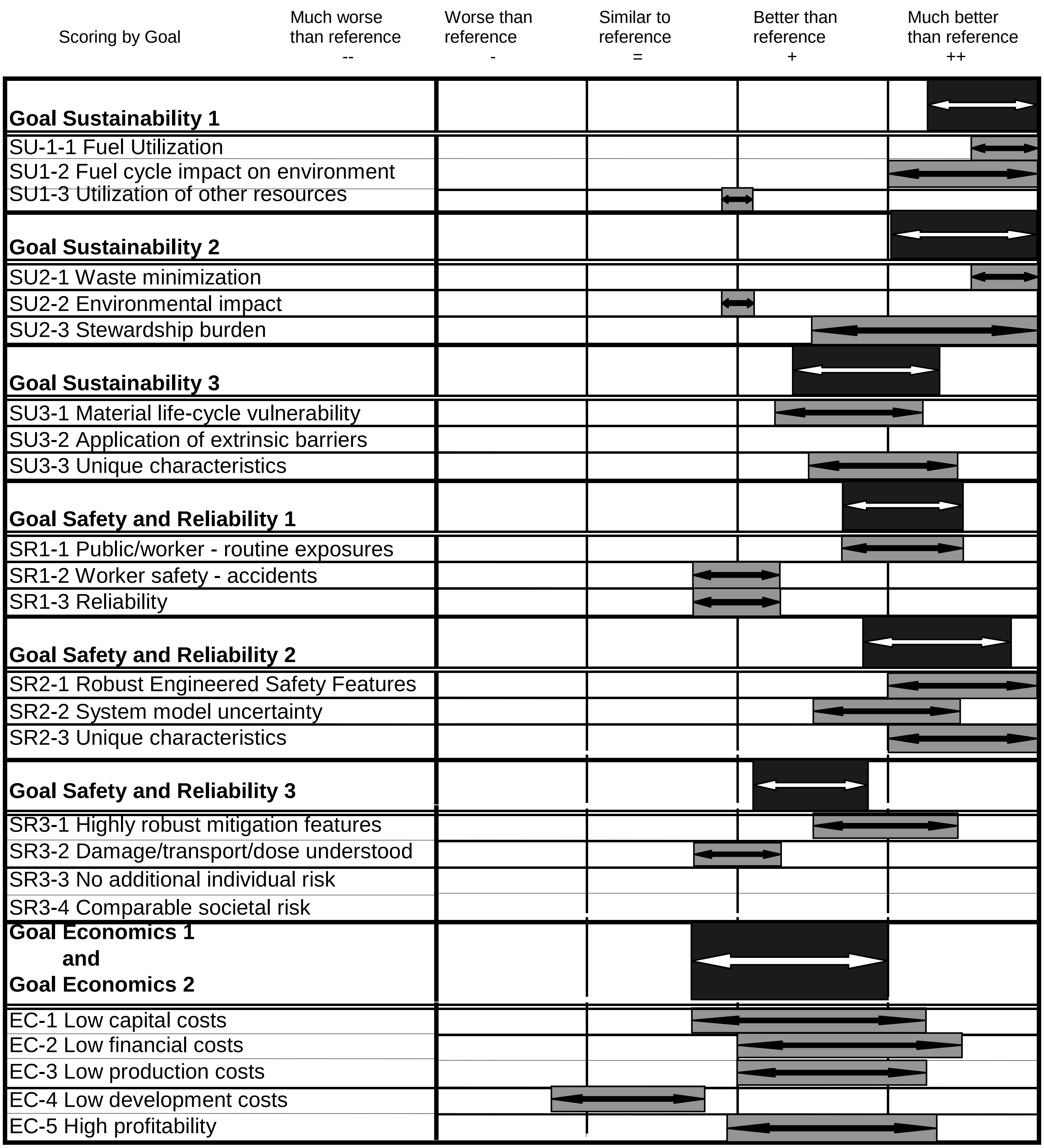
Summary Evaluation: __X__ Retain ____ Reject



Screening for Potential Scoresheet

Concept name: Medium-to-Large Metal-Fueled Sodium-Cooled Systems (Group B)

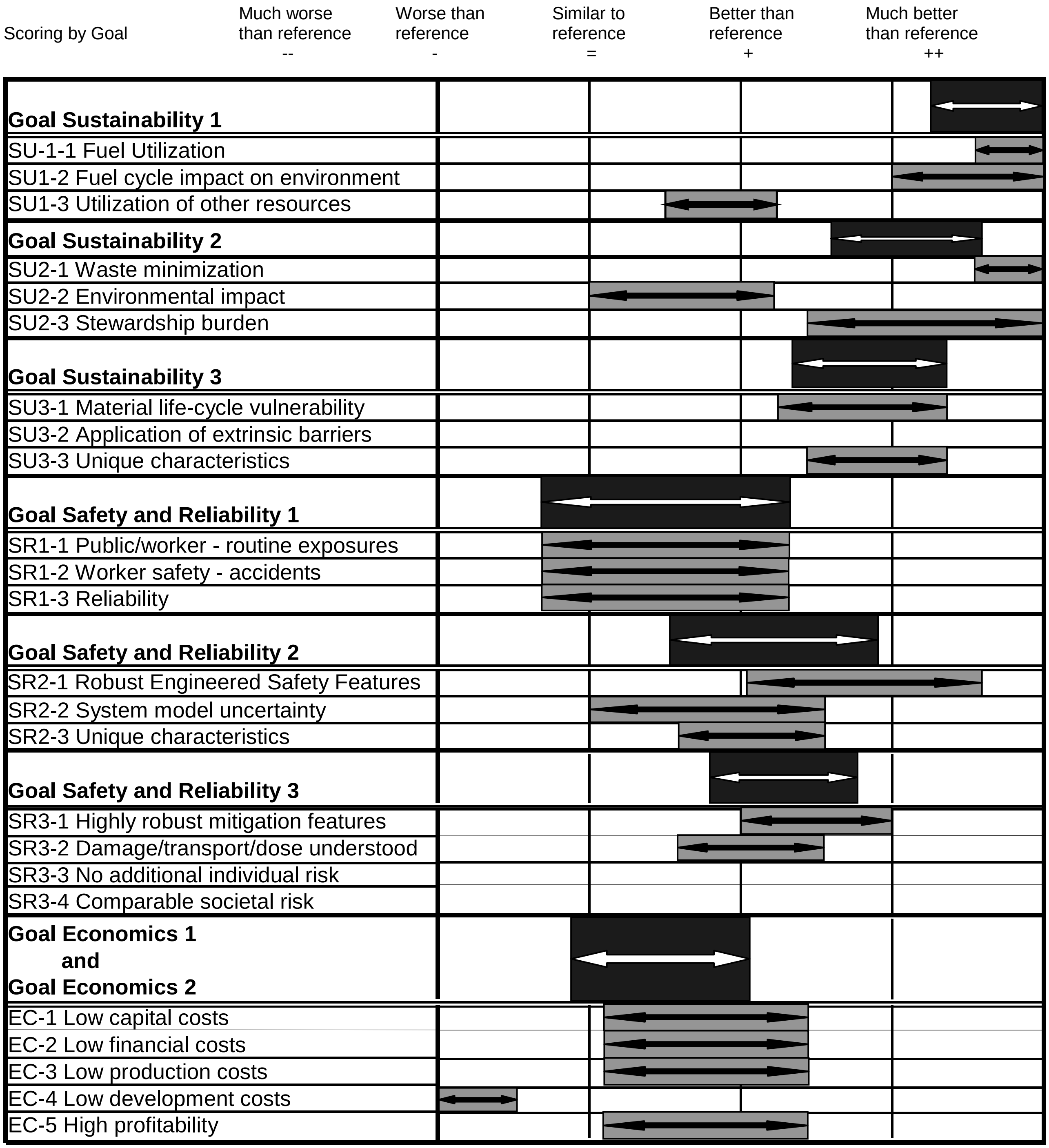
Summary Evaluation: __X__ Retain ____ Reject



Screening for Potential Scoresheet

Concept name: Medium-sized Pb and Pb/Bi Cooled Systems (Group C)

Summary Evaluation: __X__ Retain ____ Reject

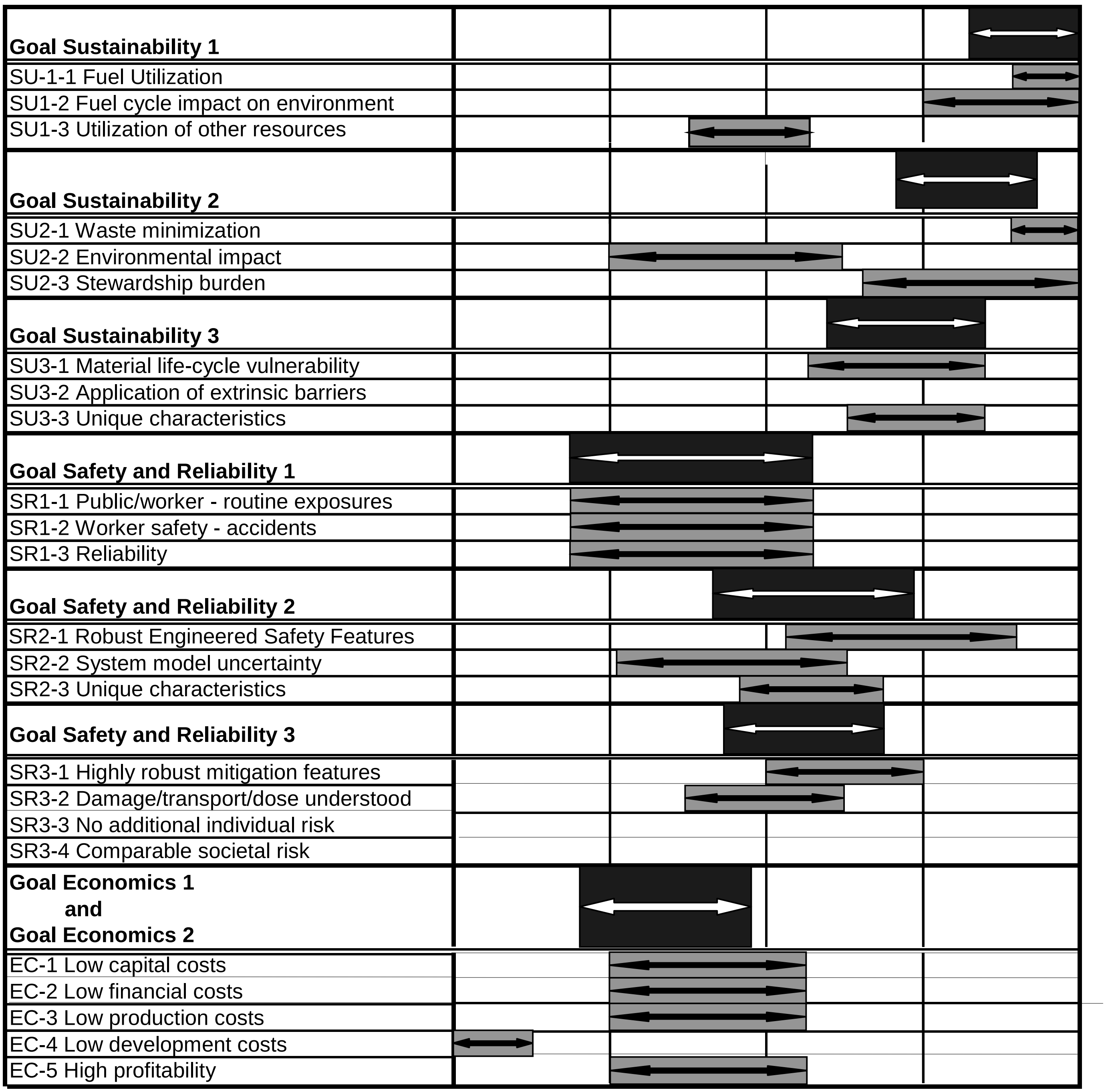


Screening for Potential Scoresheet

Concept name: Small Pb and Pb/Bi cooled systems (M11, M13, M17) and
Small Na cooled system (M24) (Group D)

Summary Evaluation: __X__ Retain ____ Reject

Scoring by Goal	Much worse than reference	Worse than reference	Similar to reference	Better than reference	Much better than reference
	--	-	=	+	++



6. SUMMARY OF LIQUID METAL REACTOR BASE TECHNOLOGY EVALUATIONS

6.1 Fuels

The development of mixed oxide fuel ($PuO_2 – UO_2$) was a cornerstone of liquid metal reactor programs around the world for over 20 years. This development culminated with the demonstration of high-burnup mixed oxide fuel in the FFTF, PHENIX, MONJU, and PFR in the United States, France, Japan, and the United Kingdom respectively. This was preceded by mixed oxide fuel testing in EBR-II, RAPSODIE, JOYO, and DFR. Nearly 378,000 oxide fuel pins have been irradiated in U.S, European, and Japanese fast reactor programs, to burnup exceeding 20% of the heavy metal. The economic incentive for lower fuel cycle costs motivated a continuous improvement in the burnup capability of mixed-oxide fuel.

In the United States, three cladding materials have been employed with mixed oxide fuel: 20% cold-worked 316 stainless steel, a modified stainless steel alloy D9 with reduced irradiation swelling characteristics, and a very low-swelling ferritic alloy HT-9. The latter exhibited no swelling due to irradiation up to a fluence of 3×10^{23} n/cm^2.

Similar alloys have been developed in Europe. Even with these improvements, the maximum fluence remains below the goal of some programs. The European Fast Reactor initiative, for example, sought a cladding fluence goal of 3.6×10^{23} n/cm^2.

There is similar pursuit of improved cladding materials in Japan, where the line of development centers around oxide dispersion-strengthened (ODS) ferritic steel. This is driven by economic incentive of obtaining higher thermal efficiencies via higher coolant core-outlet temperatures. At core-outlet coolant temperatures of 530–550°C, and cladding temperatures above 650°C, HT-9 has insufficient strength.

The response of mixed oxide fuel to off-normal events has been extensively examined in TREAT testing in the United States, and in CABRI and SCARABEE in France. These tests provided data on fuel failure mechanisms, fuel motion during failure, and coolant channel blockage. The data were then used in developing and validating fuel behavior models, transient fuel performance codes, and integrated severe accident codes.

There are few technical issues that impede deployment of mixed oxide fuel in sodium cooled fast reactors. The issue is optimization, rather than feasibility. More transient tests with advanced mixed oxide fuel pins would be a technically welcome new addition.

Metal fuel for fast reactors, although the fuel type of choice in the early fast reactor projects, fell well behind the mixed oxide until the burnup limitations were solved in the late 1960s. Metal fuels resurged with Argonne's IFR program in the mid 1980s because of simple fabrication and high thermal conductivity. Still, metal fuels are well behind mixed oxide in both static irradiation testing (~14,000 metal alloy fuel pins versus 378,000), and in transient testing. Especially in transient testing the burnup of the fuel used in the limited testing to date was relatively low (<10% burnup). Nevertheless, it may be true that experience with metal fuel is sufficient as a basis for design and licensing. Verification of the computer models and codes, though, very likely will require additional testing.

To the extent that the licensing basis for metal fuel is sufficient, it may be sufficient only for the U-Pu-Zr alloy developed in Argonne's IFR program. Numerous concepts submitted to TWG 3 involve a metal alloy fuel of Th-U-Pu-TRU-Zr, an alloy for which no data exist on fabricability, irradiation performance, or processing feasibility.

The state of development of nitride fuel is modest when compared to either the mixed oxide or the metal alloy. Nitride fuel is attractive for two reasons. It exhibits many of the same desirable characteristics of metal fuel, i.e., high heavy metal density, and good thermal conductivity. Further it has excellent compatibility with sodium (and lead). But the amount of testing to date is very small.

6.2 Coolants

Among the liquid metal reactor coolant options, nearly all of the experience worldwide has been with sodium. There has accumulated a relatively small base of information on a lead-bismuth (Pb-Bi) eutectic coolant, all of it in Russia. In recent years, there has been increasing design interest in both Pb and Pb-Bi coolant, starting with the Russian BREST reactor initiative and extending into the U.S. NERI program, among others. The U.S. Advanced Accelerator Applications program also spawned work with lead as a target/coolant. In the responses to the RFI, 14 concepts used sodium as a coolant; and 14 used Pb or Pb-Bi.

Appendix E, Section E.2, deals with these coolant options in detail, and lays out the R&D requirements.

6.2.1 Sodium

There is no question that sodium remains the coolant of choice in most fast reactor development programs and in the more mature designs. Its density, heat transfer characteristics, and compatibility with the stainless steel materials of construction render it the role of favorite among all the other possible liquid metal coolants.

It is also true that sodium as a reactor coolant has two major drawbacks: its chemical reactivity, and its positive void coefficient of reactivity in most plutonium-fueled applications. Addressing the former, sodium oxidizes rapidly in air, especially at elevated temperature. It reacts vigorously with water. The MONJU sodium leak and fire illustrates the problem, although the magnitude of the accident itself was fairly minor. There have been small sodium leaks (and small fires) at essentially every sodium-cooled reactor plant built; in some cases, several of them. These incidents, though, do not disqualify the coolant from further use.

If sodium were to boil within the core, the nuclear reactivity introduced (the “void” reactivity) is positive over a significant fraction of a plutonium-fueled core volume, and the maximum positive reactivity can reach several dollars. It turns out that in most safety analyses these coolant voiding scenarios are beyond the design base (which in the United States means a probability of less than 10^{-6} per year), but most fast reactor analysts believe that core disruptive accidents will still have to be analyzed anyway as part of a licensing process. There is, therefore, interest in reducing the void reactivity, and in developing passive means to mitigate the effects.

There are at least four important issues regarding sodium coolant:

1 How to improve the components in the primary and secondary heat transport systems to reduce cost

2 Minimization of risk from hypothetical core disruptive accidents, as noted above

3 Development of large, highly reliable steam generators

4 How to establish in-service inspection and repair techniques, particularly for in-vessel structures.

These issues form the main basis for continued R&D of sodium as a fast reactor coolant.

6.2.2 Lead or Lead-Bismuth

Pb or Pb-Bi coolant shares some characteristics with sodium that make fast reactors attractive long-term alternatives to thermal reactors. As the only alternative liquid metal coolant with significant technology development and application experience (over 80 reactor-years in Russian nuclear submarine reactors), Pb-Bi, and by extrapolation Pb, is always evaluated in comparison to sodium. This is natural and valuable, but it has limitations. Perhaps the frame of reference needs to be changed in order to do a complete and balanced evaluation. In fact, Pb or Pb-Bi coolant possess characteristics that make possible some beneficial and innovative performance goals in advanced reactor designs. They add value and diversity to a potentially long-term and large-scale nuclear power industry based on fast reactors.

Pb or Pb-Bi offer important attributes as a fast reactor coolant: they are neutronically superior to other liquid metal coolants, they are inert, and they have very high boiling temperature and low vapor pressure. The total core void coefficient is negative. These attributes offer the prospect to design a simple, low cost reactor system with enhanced safety features, possibly with a long core life.

The disadvantages of Pb or Pb-Bi coolant include its very high density, high melting point (for Pb), toxicity, requirement for corrosion protection techniques, high pumping power requirement, activation (Po-210 in Pb-Bi), activation-related contribution to the mixed waste burden, and lack of practical experience and relevant database outside of Russia. The toxicity of Pb and some of its compounds weighs negatively on public acceptance.

However, the deployment of Pb-Bi eutectic cooled reactors in the Russian nuclear submarines indicates that many of the technical problems can be overcome with adequate design, construction, and component manufacture methods. The problem of higher pumping power needs can be mitigated to some extent by spreading out the core. The higher coolant density, while imposing a burden on seismic design, offers the best natural convection potential for passive cooling and safety among all coolants.

The recent disclosure of Russian's extensive technology developments involving Pb-Bi eutectic coolant has brought new information, especially in the areas of corrosion protection and special alloys. Russia's MINATOM has been developing a Pb-cooled reactor (BREST), and IPPE (Obninsk) is promoting SVBR-75/100 reactor modules based on the submarine reactor design. There are some international R&D investments, such as within the DOE NERI program and the JNC study on fast reactor commercialization, supporting concept studies for advanced reactor systems based on Pb or Pb-Bi. Additionally, the development of accelerator-driven systems (ADS/ATW) has generated significant interest in Pb-Bi as a high power spallation target material and subcritical transmutation blanket coolant, which has led to international development programs and test facilities.

Compared to the state of sodium coolant technology, Pb-Bi and especially Pb coolant technology is still much less mature. Significant development is needed before Pb or Pb-Bi can be judged on a par with sodium as a coolant for liquid metal reactors. However, perhaps there is a need to explore the potential benefits without being confined to the conventional assumptions made for fast reactors.

Many proposed reactor concepts based on Pb or Pb-Bi coolant are innovative and target more diverse applications and markets than the conventional sodium-cooled reactors. One of the important differentiating design and deployment philosophies is that many of these concepts promote distributed energy production, as opposed to the existing large central station production model. The potential markets are more likely (but not limited to) the developing countries with little nuclear power infrastructure and expectation of rapid growth of energy consumption. It is important to investigate and verify the potential benefits of Pb or Pb-Bi coolant with some fundamental R&D and concept system studies. It is also important to verify the extensive Russian Pb-Bi nuclear coolant technology, and closely monitor the Russian development efforts.

Finally, it may turn out that the rationale for Pb or Pb-Bi coolant is not to produce electricity at the same core outlet temperature that sodium cooled systems achieve (all the concept group C systems shared this attribute). It may be that higher operating temperatures will justify lead-based coolants, and with them higher thermodynamic efficiencies and the possibility of producing non-electricity goods (e.g., hydrogen),

6.3 Fuel Cycle

6.3.1 Aqueous Recycle Technology

The PUREX process is one of the most important aqueous recycling technologies to recover U and Pu from spent fuel. It applies tri-n-butyl phosphate (TBP) as the extractant, which has the highly selective extractability of hexavalent U and tetravalent Pu, and it has already been commercialized for the processing of LWR/UO_2 and LWR/MOX spent fuels.

One of the excellent features of fast reactors is its good neutron economy. Utilizing the excess of fast neutrons enables us to construct a flexible nuclear fuel cycle system including reactors and fuel cycle facilities such that they breed or burn plutonium, transmute TRU and long-lived fission products (FPs) for reducing radiotoxicity, and enhance safety and non-proliferation features.

On the other hand, the spent fuel of fast reactors has an abundance of Pu, minor actinides (MAs) and FPs, compared to LWR spent fuel. This is because of the initial higher fissile enrichment, and because the fuel burnup of fast reactors is about 3 times as large as that of LWRs. Therefore, improvements are sought in cost, in nuclear proliferation-resistance, and in environmental impact, over the LWR recycle system.

Recently, various R&D has been carried out in order to enhance the economical competitiveness and the proliferation resistance of aqueous recycle, and to reduce its radioactive waste.

Three different options can be distinguished in the PUREX process according to the partitioning of TRUs (Np, Pu, Am, Cm) that can be achieved:

1 Standard PUREX process:

U and Pu are separated with an industrial yield close to 99.9%. MAs and FPs are conditioned in a glass matrix for interim storage and final disposal.

Improved PUREX process adapted for Np recovery:

If Np is to be co-extracted and recovered, a complete oxidation to the oxidation state Np(VI) is required. Then, the Np (VI) is extracted together with U and Pu. In this case, a recovery yield of U, Pu and Np are expected to 99.9%, 99.9% and 95-99.9%, respectively.

Extended PUREX process for MA recovery:

This process includes the separation of minor actinides (Am and Cm) and some long-lived FPs from HLLW. In this case, the goal recovery yield of U and TRUs are 99.9% altogether. For this purpose, three alternative approaches are proposed:

a. Co-extraction of minor actinides (III) and lanthanides (III), Extractants like TALSPEAK, DIDPA, TRUEX, SETFICS, TRPO, DIAMEX]

b. Separation of minor actinides in single operation leaving all the lanthanides. Extractants like TPTZ , CYANEX 301,

c. Separation of the elements contained in HLLW into four groups (Four-Group Separation Process)

The advanced aqueous recycle technology is being developed by the JNC (Japan) in pursuit of the above-mentioned objectives. In this system, Pu is extracted with Pu/Np and partitioning and purification processes are eliminated. So the system is simplified, and isolated Pu does not exist in any step of the system. In addition, the equipment is compact, and liquid waste is reduced, by recovering surplus U using the crystallization method ahead of extraction. Furthermore, TRUs are recycled to the product, which helps reduce the long-lived MAs on the outside of the nuclear fuel cycle system.

In addition, in order to improve the advanced aqueous recycle technologies further, the ion exchange method, the amine extraction method, and the supercritical fluid extraction method are also being investigated as an alternative or supplementary aqueous technology.

The major drawbacks of this technique are:

- The limited solubility of advanced fuel forms
- The limited stability of the organic extractant in high radiation fields.

6.3.2 Fabrication Technology Supporting Advanced Aqueous Recycle

Fabrication technologies are required to support the proliferation-resistant, economical fuel cycle system that is sought with advanced aqueous recycle. Of particular note, the processes all involve low fission product decontamination of the product for recycle. Also from the viewpoint of minor actinide (MA) recycle to reduce the MA amount in the high-active waste, it is important to develop the fabrication technology for MA bearing fuel.

The MA bearing and the low-decontaminated material resulting from advanced recycle have too high a radiation activity to fabricate the fuel by the traditional route of contact operations in a glove-box facility. Therefore the nuclear material must be handled in a hot-cell facility to produce fuel pellets or particles, fuel pins and finished assemblies by completely automated equipment that can be remotely maintained.

1 Fundamental technology

As noted above, the basic process technology alternatives (pelletizing and vibratory compaction) must operate automatically and be remotely maintainable in a hot-cell environment. However, it would be difficult to realize a remotely maintainable system based on a conventional pelletizing process, which consists of many precise process steps. Therefore an advanced pelletizing process with a simplified process flow is necessary.

Vibration compaction based on simple mechanisms, to produce fuel pins, has been done on a laboratory scale in several countries. Pelletizing process supporting advanced aqueous recycle

In the advanced pelletizing process, the plutonium content satisfying the fuel specification is adjusted in a mixing step of plutonium nitrate and uranium nitrate in the reprocessing plant. The mixed solution is then converted into MOX powder at a conversion facility by using the microwave direct denitration method. The MOX powder is prepared by calcination and reduction to have certain flowability that can be pelletized into green pellets, without granulation.

In order to eliminate the powder blending and granulation step from the conventional MOX pellet processes, operations of the MOX powder preparation process should be more reliable than the conventional one, because the reliability of the process dominates system throughput. The equipment has multiple functions, to convert from (uranium-plutonium-MA)-nitrate solution to MOX powder in the same container, which is then transferred to next process step on a rotating table. The equipment is based on a design philosophy of (a) minimum transfer operations of the powder to minimize dust generation, and (b) simple mechanisms in order to achieve remote-maintainability.

2 Vibration process supporting advanced aqueous/pyrochemical recycle

Gelation/vibration compaction process for aqueous recycle

Gelation processes are categorized into either external gelation or internal gelation. In both cases, the process sequence is similar. Namely, a specific organic agent is added first to an aqueous solution containing heavy metals. Then, the solution is dropped as a droplet into another solution or bath. In the course of this process, droplets become a spherical gel. In the case of sphere-compacted fuel (sphere-pac), the packing density is determined geometrically by the diametrical ratio of the spherical particles and the number of different diameters of particles. A main development item of sphere-pac is the improvement of the smear density. It is said that remote operation would be easily attained for sphere-compacted fuel fabrication because the process does not include powder-handling steps. As a result there would be no dispersion of fine powders.

Vibration compaction process for pyrochemical recycle

In oxide-electrowinning recycle, U and Pu dissolved in a chloride salt solution are precipitated at an electrode as oxides, and such oxide granules are crushed and compacted into a fuel rod. Vibropac fuel fabrication by pyroelectrochemical processing is characterized by a small number of steps due to its combination with processing, so that it would be suitable to a scale up of the facility.

3 Characteristics of the fuel fabrication system based on the technology.

The characteristics of each fuel fabrication system are evaluated briefly; see Appendix E.3.2. The advanced fuel fabrication routes such as simplified pelletizing and vibratory compaction would have the potential for reducing fuel fabrication cost and minimizing the radioactive waste generated from the fuel fabrication process.

6.3.3 Pyroprocess

In the concept submittals to TWG 3, 21 concepts of the 33 submitted referred to dry processes as either the reference or backup fuel cycle technology. In 18 of these 21 cases the specific dry process referred to was Argonne National Laboratory’s pyroprocess, and for that reason, it will be taken as the basis for discussion of “dry” or nonaqueous process technology in this report. This process involves high temperature operations involving molten salts.

In contrast to the pure extractions of conventional PUREX, the inability of the pyroprocess to recover pure fissile material is now considered an advantage with respect to proliferation resistance and is one reason for its popularity in recent years.

Compared to the conventional aqueous process, the pyroprocess has very few process steps, and the facility and equipment systems are much more compact. The recycled fuel needs to be remotely fabricated because of the inherently low decontamination factors, which is both a proliferation resistance advantage and a throughput disadvantage. The main disadvantage, though, is that the development has only reached the pilot-scale stage.

In the U.S. context, in the mid-1980s and continuing today, there is advantage to processes that can be economic at small scale, i.e., that do not depend on large economies of scale for economic competitiveness. This is a big advantage in avoiding cost penalties for fuel cycle service of the first few reactor plants deployed. In the United States it may be true that only with such an approach can initial startup deployment be contemplated. The pyroprocess has that potential.

The key step is electrorefining. The molten salt medium for electrorefining is a solution of LiCl-KCl eutectic and dissolved actinide chlorides, such as UCl_3. The operating temperature is 500°C. With this system, chopped spent fuel is loaded into the electrorefiner in baskets. The fuel is electrochemically dissolved into the system in an operation in which the baskets are the anodes and another electrode in the salt phase is the cathode. Uranium with little TRU material can be collected on steel electrodes (solid cathodes), and TRU materials can be co-deposited with uranium in liquid-cadmium cathodes. Because of the chemical activities of the TRU elements in cadmium, they can be easily deposited with uranium in liquid-cadmium cathodes but not on solid cathodes. The cathode products from electrorefining operations are further processed to distill adhering salt and cadmium and to consolidate the recovered actinides. The recovered actinides are remotely fabricated into new fuel for recycle by injection casting of slugs from 34–45 cm in length, which has been used to produce over 200,000 metal fuel elements for EBR-II, 35,000 of them remotely. An alternative casting process, centrifugal casting, has been studied in Japan.

The alkali, alkaline earth, rare earth, and halide fission products are primarily in the salt phase. The elements that distribute into the salt phase are eventually disposed in a ceramic high level waste. The noble metal fission products and zirconium alloying materials are distributed between the fuel cladding and the interior of the electrorefiner in the anode baskets. The cladding hull segments and the retained fission products are eventually stabilized into a metal high-level waste.

The IFR program was terminated in 1994 prior to demonstrating the technology through the recycle of spent fuel from the Experimental Breeder Reactor II (EBR-II). When this program was terminated, the pyroprocess was modified for the treatment of the EBR-II fuel for eventual disposal. The key difference between the use of the technology for fuel treatment versus fuel recycle is that the transuranics are not recovered for fuel treatment. They are instead allowed to build up in the electrorefiner salt phase and then eventually disposed of in the resulting ceramic high-level waste.

The spent fuel treatment technology was successfully demonstrated with EBR-II fuel. During this demonstration by Argonne conducted between June 1996 and August 1999, 100 EBR-II driver (400 kg highly-enriched uranium) and 13 EBR-II blankets (600 kg depleted uranium) assemblies were treated. DOE decided to use this technology to process the remaining EBR-II fuel (approximately 25 tonnes) and some sodium-bonded metal fuel from the Fast Flux Test Facility.

The Spent Fuel Treatment Program at Argonne demonstrated many parts of the pyroprocess fuel cycle, but there are still key aspects that have yet to be demonstrated on a large scale with radioactive materials. The main outstanding issues and opportunities:

- Recovery of transuranics
- Use of the test facilities as a test bed for transparent safeguards development
- Irradiation of remotely recycled fuel to prove performance
- Materials R&D to decrease process losses to secondary streams
- Demonstration of reduction of HLW volumes through zeolite ion-exchange processes
- Continued development of adaptations for oxide and nitride fuels.

7. CONCLUSIONS

In this preliminary round, 33 liquid metal reactor concepts were screened for their potential to meet the Gen IV goals. Four were rejected, and 29 were gathered into five groups and one stand-alone concept and "screened in," i.e., passed to the next round of evaluation. It is quite possible that the TWG will assign the stand-alone concept, M2, to concept group D in the next round. Similarly, concept group E which aims to eliminate the IHTS, may be incorporated as an optional R&D path associated with both concept group A and group B.

Although a useful start was made in this preliminary round on identification of R&D needed to bring the concept groups to a deployment (or serious development) state, considerably more work needs to be done to put these R&D requirements on a more equal footing, with a more uniform approach. This will be done in the coming months.

Appendix A

Concept Group A

Medium-To-Large Sodium-Cooled Systems

Appendix A

Medium-to-Large Sodium-Cooled Systems

Jean-Louis Carbonnier, Masakazu Ichimiya, John C. Lee

A.1. Introduction

We have reviewed and evaluated five Generation IV sodium-cooled reactor concepts with medium-to-large power ratings. Out of a collection of liquid-metal cooled reactor (LMR) concepts under review by TWG 3, Group A includes several mature concepts, certainly in terms of the coolant technology. The five concepts represent advanced reactor designs developed by four countries: Japan, Russia, France, and the United Kingdom.

As is characteristic of LMR designs in general, all Group A concepts operate at low primary system pressures and yield fast neutron flux spectra. These basic characteristics translate into plant features that would qualify them as nearer-term Generation IV concepts, including (a) enhanced safety through passive shutdown heat removal, (b) multiple recycling of transuranics (TRUs) from spent fuel, (c) high thermal efficiency, and (d) potential for fissile breeding. Some designs feature a pool-type primary system, while some others utilize a loop-type heat transfer system. Efforts have been made in all Group A concepts to address some of the issues that have plagued the development of LMRs over the past few decades. In particular, special attention has been given to the potential for positive sodium void reactivity (SVR) feedback in loss-of-flow (LOF) events, to potential sodium-water reactions in case of sodium leakage, and to in-service inspection and repair capabilities. Effort has been made also to reduce or eliminate, through varying designs, the economic penalty associated with the intermediate heat exchanger (IHX).

A.2. Concept Description and Comparison

To facilitate succinct evaluation of the concepts, we compare key system characteristics for all five concepts in Table A-1. The comparison focuses on overall plant characteristics, core and fuel characteristics, and safety features, with additional considerations given for materials, economics, and R&D needs identified by the sponsoring organizations. A brief description of each of the concepts is provided in this section, with the information in Table A-1 utilized to the fullest extent possible. It should be mentioned here that the information for concept M6 has been collected and summarized by a member (J. C. Lee) of TWG 3 and is likely incomplete and not current in many respects.

Some of the advanced safety features common to the concepts reviewed include:

- Diversity in the reactor control system (RCS), utilizing passive modes of insertion
- Diversity and redundancy in the decay heat removal system (DHRS).

All of these features are expected to decrease the risk of operation of these systems compared with current light water reactor (LWR) plants. Various additional safety features are included in some designs to reduce the residual risk associated with extremely low probability recriticality events that could lead, ultimately, to core disruptive accidents (CDAs).

The reliability of LMR plants is expected to eventually become comparable to the very high levels of reliability of the current fleet of LWR plants.

It should be noted here that the fuel composition for several concepts is presented as either mixed oxide (MOX) or UO_2, but it is generally understood that any of the designs could be changed to metal fuel, with due considerations given for the overall system performance. Likewise, any of the designs could operate as a breeder or burner. We recognize that a number of specific data are often unavailable for detailed comparison at this point. Nonetheless, it may be useful to note a number of similarities among the concepts, e.g., for all five designs where the data are available, (a) the core outlet sodium temperature is around 820 K, (b) the thermal efficiency is estimated to be around 40%, and (c) the core height is chosen as 1.0 m.

A.2.1 Concept M4, JNC Sodium Cooled Fast Reactor, JSFR

JSFR is a loop-type reactor that inherits Japanese fast reactor technologies and experience, including those associated with JOYO and MONJU. For economic competitiveness, the design offers the following key features:

- Simplified and compact structure of the reactor
- Shortened piping layout
- Two loops for the cooling and heat transfer system
- Integration of the IHX and primary pump.

These cost reduction measures benefit from advanced technologies, such as a three-dimensional seismic isolation system, 12-Cr steel, advanced structural design standards, and a core design free from recriticality accidents.

With high priorities given to ensuring safe operation of the plant, the following safety features were introduced:

- Reactor shutdown system with built-in redundancy and diversity
- Passive shutdown capability through the self-actuating shutdown system (SASS)
- DHRS with natural circulation capability
- Primary and secondary systems fully enclosed by a guard vessel
- Reinforced water leak detection system for the steam generator.

Furthermore, special effort has been made to meet the safety requirements, which preclude recriticality accidents even in case of a postulated CDA. In order to prevent prompt recriticality through the initiating and transient phases of representative CDAs, the SVR is limited to 6 dollars or less and the fuel assembly with inner duct structure, the FAIDUS concept, is adopted to facilitate molten fuel discharge from the core region. The effectiveness of the FAIDUS concept is now under investigation by performing both out-of-pile and in-pile experiments.

Achieving an efficient use of natural resources, reduction of environmental burden, and proliferation resistance was given a high priority in the JSFR core design. Effort is underway to evaluate the impact of recycle technology in the overall environmental and sustainability considerations for the JSFR design.

A.2.2 Concept M5, Modular Type JNC Sodium Cooled Fast Reactor, M-JSFR

M-JSFR is a loop-type medium-size reactor, featuring basic reactor concepts similar to the full-size JSFR concept discussed above, but offers the following advantages of a modular design:

- Flexibility in meeting various levels of power requirements of the utility companies
- Reduction of development risk compared to full-size reactors.

On the other hand, as a medium-size reactor, M-JSFR suffers from an economic disadvantage in construction cost per unit electric power. To compensate for this disadvantage, advanced modular design features are adopted to improve the economic competitiveness of the concept, together with the following specific cost reduction measures:

- Simplification of the secondary and tertiary cooling systems, featuring one SG per reactor and one turbine-generator for three reactors
- Simplification of the DHRS, through enhanced natural circulation capability
- Adoption of a straight-tube SG design.

The M-JSFR design shares with JSFR the same safety design principle and the reactor core design principle, maintaining the goal of achieving an efficient use of natural resources, reduction of environmental burden, and proliferation resistance.

A.2.3 Concept M6, BN-800

This concept utilizes a pool-type fast-spectrum reactor core design, which incorporates enhancements to the BN-600 reactor that has been operating since 1980 at the Beloyarsk Nuclear Power Plant (NPP) in Russia. Based on the successful sodium-cooled LMR experience with the loop-type BN-350 design and the pool-type BN-600 design, the construction of the BN-800 plants was initiated in 1986 both as Beloyarsk Unit 4 and as South Urals Units 1 and 2. The construction was stopped in 1990, following the Chernobyl accident, and the completion of the BN-800 plants has apparently been put on hold indefinitely. Key features of BN-800 include:

- Double-walled reactor vessel housing the core and IHX
- Elimination of positive SVR through a sodium plenum
- Double monitoring of sodium leak and fire
- Active and passive means to eliminate sodium leaks and suppress sodium fires.

Special effort has been made to reduce the potential for positive SVR by introducing a sodium plenum in the top axial blanket. This will allow the voiding of sodium initiated near the top of the core to propagate into the sodium plenum, thereby increasing the neutron leakage out of the core and resulting in negative reactivity feedback. The sodium plenum concept has been tested experimentally at the Institute of Physics and Power Engineering at Obninsk. This includes both a full-scale reactor model and simplified mockups. This experimental program provided valuable data for validating sodium-reactivity calculational models in Russia and other countries. A seven-nation (Germany, France, UK, Italy, Japan, India, and Russia) team performed a comprehensive transient and accident analysis for BN-800 recently under the aegis of the International Atomic Energy Agency.

A.2.4 Concept M22, Sodium Cooled Liquid Metal Reactor, RNR 1500

This concept, based on the solid European expertise in sodium-cooled fast reactors, is a pool-type fast-spectrum large-size reactor core design with a power rating of 1.5 GWe. The design incorporates enhancements to the industrial power plant Superphénix (1.2 GWe) in order to improve safety and cost competitiveness compared to future LWRs.

The concept is based on a safety design making extensive uses of the defense-in-depth principle and special efforts have been made at three different levels:

- Risk prevention, achieved through the selection of appropriate materials, strict application of procedures, and provision of diverse and redundant systems for failure detection and reactor protection
- Risk minimization, through enhanced retention capability of the containment, and enhanced reliability of shutdown and heat decay removal systems
- In-service inspection capabilities through improved visual inspection techniques for internal structures and the means of access provided by design.

Particular effort has also been made to optimize the reactor design from the viewpoint of general structural design simplification and compaction (notably for core support structures and once-through straight-tube bundle SGs). This provides improved cost competitiveness for a large integrated concept, together with provisions made for convenient in-service inspections and repairs.

The core concept has been designed to provide breeding/burning flexibility and TRU incineration capabilities based on a MOX fuel cycle.

A.2.5 Concept M30, CPFR

This concept is a large (1.5 Gwe, or twin 0.8 Gwe units) pool-type MOX-fueled reactor. One objective is low capital cost due to a reduction in the number of reactor primary circuits, a smaller primary vessel and hence, a smaller reactor building.

The safety approach approval is said to be based on “conventional LMFBR safety features.” In addition to two separate and diverse absorber systems for reactor shutdown, a self-actuating shutdown system is provided. Although no details are provided, decay heat removal is said to be accomplished by decay heat exchanger units immersed in the primary pool, and by vessel wall cooling.

A.3. Potential of the Concept for Meeting the Generation IV Goals

A.3.1 Evaluation against Criteria/Metrics

A.3.1.1 Sustainability

As is the case in general for fast-spectrum LMR systems, the concepts in Group A are expected to accommodate multiple recycling of the entire TRU inventory, and could operate as a breeder. These features offer significant advantages over current LWR systems for fuel utilization and waste minimization. With respect to environmental impact and proliferation-resistance, many details are missing for an in-depth evaluation. Nevertheless, with the fuel processing technique cited by most concepts, advanced aqueous recycle, it is intended that uranium and plutonium are co-extracted, along with most of the minor actinides. Moreover, the production and utilization of plutonium can be balanced by varying the breeding or conversion ratio. These are considered assets with respect to proliferation resistance. From an environmental perspective, uranium mining, milling, and enrichment are avoided for a very long time (the latter forever) through use of the huge stocks of depleted uranium already available. Also, if most of the actinides are removed from the HLW and recycled, the technical requirements for isolation placed upon the repositories are eased. All these factors augur well for sustainability.

A.3.1.2 Safety and Reliability

The advanced safety features discussed in Section A.2 offer the potential for increased passive safety, compared with current LWR plants. In spite of the significant effort made to eliminate or reduce positive SVR feedback, it has not been possible thus far to completely eliminate the potential for secondary criticality in LMRs (we acknowledge in this regard, the efforts in Japan to develop the recriticality-free concept embedded in concepts M4 and M5). Thus, the residual accident probability for recriticality and CDAs has to be evaluated and will likely be a significant factor in the licensing and design certification process in all participating countries. Despite some incidents associated with sodium usage, including those at Superphénix and MONJU, important experience has been accumulated from EBR-II, FFTF, Phénix, JOYO and BN-600, among others. This allows us to offer reasonable confidence in achieving safe and reliable operation of the plants that meet the safety and design specifications of the concepts evaluated here.

A.3.1.3 Economics

As discussed in Section A.2, significant effort has been made in all five concepts reviewed here to simplify and optimize the design, e.g., consolidating the primary coolant pump with IHX. See Section 3.1.5 for the concept group E discussion, and bold proposals for elimination of the IHTS (the secondary sodium system). Nonetheless, there exists considerable uncertainty if any of the designs will be competitive economically with Generation III systems, including AP600, System 80+, and the Advanced Boiling Water Reactor, both in terms of the overall financial risk associated with the construction cost and eventually the generation cost. Because of the medium-to-large power ratings, Group A concepts may offer some advantage in the overall capital and generation costs per kWe. This may in particular be the case for concepts M4 and M5, where specific effort has been made to implement simplified and compact structures throughout the plant.

A.3.2 Summary of Concept Potential

Strengths: The concepts offer enhanced safety features, increased fuel utilization, and waste recycling and transmutation.

Weaknesses: In spite of valuable operating experience at EBR-II, FFTF, Phénix, JOYO, BN-600, and elsewhere experience in the sodium technology is still limited and the possibility of recriticality will likely remain as a safety or licensing issue, although the latter is being investigated. Proliferation issues have to be resolved at the national and international level before recycling may be accepted in all participating countries. Economic projections will require extensive confirmation efforts.

A.4. Technical Uncertainties

We believe the R&D needs included in Table A-1 represent rather a preliminary compilation of areas that require substantial investment in the near future. As the designs undergo the actual licensing and design certification process, a number of additional R&D needs will undoubtedly be identified. This issue will require further discussion among TWG 3 members as part of the continuing evaluation process.

A.5. Overall Concept Potential vs. R&D Risk

As some of the more mature LMR concepts, Group A concepts entail the least R&D risk and should be accepted for more detailed analysis and evaluation. Specific potential for meeting various Generation IV goals are indicated in the scoresheet (Section 3.2), where comparison is made with Generation III systems whenever possible. In this regard, we assume that a number of enhancements proposed for Group A concepts would be realized through Generation IV R&D effort.

Table A-1. Comparison of key system characteristics for medium-to-large sodium-cooled systems.

Concept Number and Name	M4 JSFR	M5 M-JSFR	M6 BN-800	M22 RNR 1500	M30 CPFR
Sponsoring Organization	Japan Nuclear Cycle Development Institute, Japan		Beloyasrsk NPP, Russia	CEA, France	NNC Limited, UK
1. Overall Plant Characteristics					
Power output (GWe/GWt)	1.5/3.57	0.5/1.19	0.8/2.1	1.47/3.6	1.5/3.6
Thermal efficiency (%)	42	42	38	41	42
Number of primary loops	2	2	3	3	2
Primary loop arrangement	loop-type, RCP and IHX integrated		pool-type	pool-type, 3 RCP, 6 IHX	2 RCP
Reactor vessel arrangement	Double reactor vessel		double reactor vessel, core catcher	double vessel, anchored to concrete vault	molten fuel debris tray
Reactor vessel height/diameter (m)	18.8/9.6	18.5/6.5		16/17.2	
Primary/secondary/tertiary coolant	$Na/Na/H_2O$		$Na/Na/H_2O$	$Na/Na/H_2O$	
Secondary loop or power generation features	1 NSSS per turbine	2 loops/SG, 3 NSSS per turbine	1 NSSS per turbine	6 independent Na loop/IHX	4 IHX and 2 SG
2. Core and Fuel Characteristics					
Core height/diameter (m)	0.8-1.0/5.0	0.8-1.15/2.6	1.0/2.45	1.0/2.03	1.0/?
Heat generation rate	43 kW/M (MOX fuel)			41-52 KW/m	
Core outlet Na temperature (K)	823		820	818	
Core Na temperature rise (K)	155		193	150	
Core outlet pressure (kPa)	200		54		
Fuel composition and configuration	MOX or metal, fuel assembly with inner duct structure (FAIDUS), homogeneous or heterogeneous core layout		UO_2, MOX, metal; annular pellet	MOX, annular pellet	MOX, annular pellet
Fuel fabrication and processing	Vibropacking and injection casting; aqueous or non-aqueous				
Breeding ratio	1.1-1.3	1.2	1.0-1.27	> 1.02	1.2
Average discharge fuel burnup (MWd/kgHM)	150		100	145	180
Fuel cycle and fuel utilization	30-40 year doubling time, 17-20 month cycle, 4 batches			1700 EFPD, 5 batches	
3. Safety Features					

Table A-1. (continued).

Concept Number and Name	M4 JSFR	M5 M-JSFR	M6 BN-800	M22 RNR 1500	M30 CPFR
Special features in reactor control system	Active RCS with redundancy and diversity, SASS for passive shutdown using Curie point alloy		hydraulic rod insertion	3 RCS; temperature triggered rod insertion	diverse self-actuating shutdown rods
Decay heat removal system	Active and passive DHRS		active and passive DHRS	active and passive DHRS	
	IRACS and DRACS	PRACS and IRACS			
Reactivity coefficient and severe accidents	SVR < 6 dollars, molten fuel discharge through FAIDUS, precluding recriticality		SVR < 0 through Na plenum	small SVR, thick main vessel	small SVR, annular pellet, debris tray
Piping arrangement	Double shell, inert gas filled		double shell		
In-service inspection and repair	Monitoring of Na leak in reactor vessel and piping, inspection of SG tubes		monitoring of Na leak and fire	main/guard vessel inspection; system failure detection	
Other safety features	Passive decay heat removal through natural circulation		active control of Na fire, RVACS	air cooling of steam generator	
4. Materials Considerations					
Fuel, piping, vessel	ODS for fuel cladding. High-Cr steel for cooling system			AIM1 for fuel cladding	PE16 for fuel cladding
5. Economics					
Capital cost	< 2/3 of LWR cost, 46-month construction period		parity with LWR aimed	parity with PWR aimed	
Generation cost	Parity with LWR expected in the future, availability = 92%				
6. Special Features					
Waste management, safety feature	TRU with I-129 and Tc-99 transmuted		Na plenum design tested	minor actinide transmutation studied	
7. R&D Needs					
Safety	Demonstration of SASS and FAIDUS, development of completely passive DHRS			in-service inspection and repair	
Fuel cycle	Demonstration of isotope separation technology, development of metal fuel fabrication and recycle technology			spent fuel recycle	

See list of acronyms at beginning of report
Blanks indicate only that no information was developed in this work.

Appendix B

Medium-To-Large Metal-Fueled Sodium-Cooled Systems

Appendix B

Medium-to-Large Metal-Fueled Sodium-Cooled Systems

C. E. Boardman, D. H. Hahn, D.C. Wade

B.1.1 Introduction

The Group B concepts were sponsored by General Electric and Lawrence Livermore National Laboratory (M1 and M25), the Japan Nuclear Cycle Development Institute (M4 and M5), the Korean Atomic Energy Research Institute (M7and M8), Argonne National Laboratory (M15), and the Toshiba Corporation (M33). A summary of their major features is provided in Table B-1.

Seven of the eight concepts in the Group B category are medium-sized sodium cooled fast reactors with metal alloy fuel. The lone exception is M4, which is a large monolithic reactor with a thermal rating that is significantly greater than the other concepts in Group B. Both M4 and M5 are included in concept group A. They are also included here because metal fuel remains an option under consideration in Japan.

The motivation for the six medium size modular concepts are associated with the desire to utilize the advantages of factory fabrication, improved plant availability, and a shorter construction schedule to overcome the disadvantages of their relatively small size and lack of "economy of scale". Clearly all economies of scale factors work to the advantage of the monolithic plants. The relatively small size of the modular concepts makes it feasible to use simple fully passive decay heat removal systems that simplify the design and improve the competitive potential of the concepts.

A potential advantage of modular concepts is that they can be commercialized at a lower development cost than a large monolithic plant. This advantage stems from the fact that the need for component scale-up is eliminated and the design can be certified through the construction and testing of a single reactor module, roughly a 350-MWe system.

Further advantage of the modular concepts is that they might be able to use a "license by test" approach.[1] This means that the initial zero and low-power testing of the prototype would be used to confirm the thermal-hydraulic performance of the reactor system and key components. And the safety test program would be used to demonstrate the other key features such as the ability to accommodate ATWS events. For example, the single module prototype test can be used to determine and demonstrate thermal capability of the passive shutdown heat removal systems and the reliability of the fuel. The higher availability and simpler operational requirements of a modular prototype makes it an ideal machine for use in developing advanced fuels. For example, it might be expected that prototype testing will confirm that HT9M clad metal fuel can achieve burnups and fluences in excess of 20 at.% and 3.8E23 n/cm2, respectively, but of course burnups to this level must be achieved incrementally, as the prototype reactor operation proceeds. The prototype tests will be used to demonstrate the capability of the fuel and reactor system as required by the NRC for Design Certification.

B 1.2 Concept Description

Table B-1 at end of this Appendix summarizes the major design features and performance characteristics of the reactor concepts in Group B. Appendix E discusses the key base technologies utilized by the concepts.

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1. It must be noted that monolithic plants can adopt much the same strategy through development of "prototype" plants of a lesser size, but with the same essential features, of the larger plant design.

All of the concepts in Group B have adopted metal fuel that is expected to have superior performance characteristics, including:

- Metal fuel is denser and yields a harder neutron spectrum than oxide fuel.
- It is possible to reach conversion ratios in excess of 1.3, if desired.
- A smaller core volume is required.
- A near zero cycle burnup reactivity swing can be obtained. This reduces the worth of the overall control system and the individual rods. This reduces the severity of a control runout accident and it also reduces the sensitivity of the reactor system to seismic events.
- Axial blankets are not required to achieve a conversion ratio of one.
- Sodium bonded metal fuel is fully compatible with the coolant. Run Beyond Cladding Breach (RBCB) is possible, as demonstrated by EBR-II.
- High thermal conductivity reduces the operating temperature of the fuel.
- Lower Doppler feedback reduces the number of gas expansion modules (GEMS) required to passively accommodate unprotected loss of flow events.
- The energetic potential of metal fuel is very low (near zero) due to its low melting temperature and its propensity to expand in the axial direction during postulated overpower events.
- The potential for a re-criticality event following a postulated HCDA or an over temperature event is expected to be significantly reduced with metal fuel. The lower melting temperature and self-heating of the fuel gives it the ability to drain into the high-pressure inlet plenum where it will be retained in a non-critical condition. Similarly, the measures employed in Japan's "re-criticality free" approach, embodied in concepts M4 and M5, aim to reduce or eliminate the potential for recriticality.
- Pyroprocessing of metal fuel is highly diversion resistant, compact, less complex, less costly and has fewer waste streams than conventional aqueous (PUREX) process used for oxide fuel.

B 1.3 Potential of Concept for Meeting the Generation IV Goals

B 1.3.1 Evaluation Against Criteria/Metrics

B 1.3.1.1 Sustainability

Except for process losses, 100% of the uranium introduced into the fuel cycle can be utilized to produce energy in most fast reactors, and this is the case for all the Group B concepts except concept M8, which utilizes an ultra-long-life core that would be remotely re-clad in order to allow the fuel to reach an extremely high burnup prior to disposing of the spent fuel. All other concepts in Group B (and in groups A, C and D as well) could utilize the present U.S. stockpiles of depleted uranium to fill, for example, all U.S. electrical requirements for the next 1500 years, without the need for additional mining, milling, or enrichment operations. Because the minor actinides are not separated from the Pu but are utilized in the creation of new fuel pins, the fuel is always intensely radioactive and must be remotely fabricated in inerted, heavily shielded hot cells.

During recycle, the fission products are separated and conditioned for disposal prior to their shipment to an off-site repository. Due to the shorter half-life of the FPs compared to the heavier actinides that are recycled and burned, the effective repository mission is reduced from millions of years to 500–1000 years, depending on what fraction of the actinides are removed from the HLW and recycled.

The fuel cycle employed by the Group B concepts provides an economical long-term source of electrical energy while reducing proliferation risk by allowing the production and utilization of Pu to be balanced by varying the conversion ratio. Studies have shown that a conversion ratio can be varied from 0.8 to more than 1.3. These concepts also reduce the need for additional enrichment facilities that would be required to extract U-235 for use in conventional thermal reactors.

B 1.3.1.2 Safety and Reliability

The use of passive shutdown, passive shutdown heat removal, and passive post accident cooling (containment or in-vessel) is used to (1) simplify the plant, (2) eliminate the need for operator action to protect the plant investment, and (3) reduce the cost of maintaining the plant. The reduced risk and radiological release characteristics are expected to allow the Group B concepts to meet the Low Level Planning Action Guidelines limits, eliminating the need for formal evacuation planning. Some concepts in Group B have elected to use well-demonstrated, redundant, and diverse active safety systems. An example is the use of piped (loop) decay heat removal systems that utilize pumps and fans to reach rated capability but can also operate in a passive natural circulating mode at reduced capacity. Both the fully passive systems and the active systems, with partial capability in the passive mode, are expected to be licensable.

The metal-fueled Group B concepts are designed to accommodate a set of accident initiators that in prior LMRs led to coolant boiling and core melting. These “accommodated” Anticipated Transients Without Scram (ATWS) bound the most probable core disruptive accident initiators.

One of the Group B concepts (M1) is developed by extrapolating design concepts of the early PRISM initiative (@ 125 Mwe), that was reviewed by the US NRC (NUREG-1368) and determined to be “*licensable to the best extent possible for the current state of the design detail.*”

B 1.3.1.3 Economics

The economic performance of several of the Group B concepts (M1, M4, M5) has been extensively evaluated,[1] based on information provided by the respective sponsors. The economic performance of these concepts is claimed to be fully compatible with the Generation IV goals, as discussed below. While the other concepts within Group B are not as fully developed, they are also claimed to be economically viable when fully developed.

For example, the generation cost of M4 is estimated to be at parity with future LWRs. For M1, the NOAK two power block plant rated at 1520 MWe (net) is claimed to have a levelized busbar cost of 29 mills/kW-hr. This estimate is based on detailed ALMR commodities estimates generated during the ALMR program that is said to be supported by a $100 million database. The breakdown of this estimate is as follows:

Capital Cost	16.8
O&M Cost	6.2
Fuel Cycle Cost	5.0
Decommissioning Cost	1.0
Total Busbar Cost	29.0 mills/kW-hr

A 29 mills/kW-hr generation cost, if achieved, would be competitive with other generating systems presently in use.

1. By others, not by TWG 3

B 1.4 Technical Uncertainties

The Group B concepts have very similar development and R&D needs. The reactor design tasks are relatively straightforward component development tasks. There is perhaps only modest risk that the R&D tasks will not be successfully carried out. This is consistent with the fact that the world has already constructed more than 20 fast reactors, the largest being the 1200-MWe French Superphenix plant, depicted in Table B-2.

The challenge is not to just to build another demonstration plant; but to design, construct, and demonstrate the operational viability of one that is economically competitive. Previous demonstration plants were designed and built in a period in which relatively little attention was paid to minimizing the construction cost. At that time, it was believed that the continuing escalation in the cost of uranium and fossil fuels would make the fast reactor necessary even if its capital cost was considerably higher than competing LWRs. However, in recent years it has been recognized that in order to justify its development, the fast reactor must show significant competitive potential. Major reductions in the commodities and capital cost are claimed in the latest advanced fast reactor designs. The latest European and Japanese fast reactor plant designs, such as the 1450-MWe EFR and modular and monolithic designs by JNC in Japan (M4 and M5), represent very serious efforts directed toward designing a competitive fast reactor. For M4, the capital cost is estimated to be 2/3 of current LWR capital cost. Thus, most of the R&D tasks are directed to ensuring that the plants are not only safe, reliable, and licensable but are also competitive.

Key R&D tasks involve:

- Fuels (see Section E.1 of Appendix E)
    - Validation of performance characteristics
    - Validation of behavior under accident conditions
    - Waste and Recycling (see Section E.3 of Appendix E)
    - Demonstrate acceptable decontamination factors, low heavy metal losses, waste treatment method, and economics of operation
- Safety Methods
    - Containment analysis following Design- and Beyond-Design Basis Accidents
    - Core energetic (HCDA) accidents
    - Steam sodium water reaction analysis
    - PRA analysis.

B 1.5 Statement of Overall Concept Potential versus R&D Risk

All concepts in Group B are expected to meet the Generation IV goals with modest risk so that the reactor systems will meet all of its operational goals as illustrated in the Screening for Potential Score Sheet (see Section 5.2). The key R&D task that must be completed for all of the concepts is a long life metal fuel and its associated fuel cycle including processing, waste treatment, and a high level of intrinsic proliferation resistance as discussed in the separate Base Technology Evaluations (Appendix E).

The one exception is concept M8, which will not match the sustainability or uranium utilization levels that the other concepts in TWG 3 can achieve through full recycle. Concept M8 is designed to achieve a very high burn up through the use of re-cladding operations. However, concept M8 utilizes a once-through fuel cycle that will end up sending highly enriched spent fuel to the repository.

Group B concepts should be accepted for continued evaluation.

Table B-1. Major design features and performance characteristics.

Feature	M1/M25	M4	M5	M7	M8	M15	M33
Overall Plant	**S-PRISM Modular**	**JSFR Monolithic**	**JFSR Modular**	**KALIMER Modular**	**ULLC w/ Re-cladding**	**AFR-300 Modular**	**Self Consistent Core**
- Net Electrical Output, MWe	2280	2856	2856	150	500	300	640
- Net Station Efficiency, %	38	40	40	38.2	36.3	~ 38	40
- Number of Power Blocks	3	2	2	1	1	1	NA
- Number of Reactors/Power Block	2	1	3	1	1	1	NA
- Expected Plant Capacity Factor, %	93	92	92	Not Available	Not Available	Not Available	Not Available
Power Block							
- Number of Reactors	2	1	3	1	1	1	NA
- Net Electrical Output, Mwe (gross)	825	1500	1500	150	500	300	NA
- Steam Generators	Two	2	3	2	NA	NA	NA
- SG Type	Helical Coil	Helical Coil	Straight Tube	Helical Coil	NA	NA	NA
- Steam Cycle	Superheat	Superheat	Superheat	Superheat	NA	NA	NA
- Turbine Type	TC-4F	NA	NA	NA	NA	NA	NA
- Turbine Throttle Conditions, C/ MPa	468/17.3	495/16.67	495/16.67	483/15.5	NA	NA	NA
- Feedwater Temperature, C	201	233	240	NA	NA	NA	NA
Reactor System							
- Thermal/Electric Power, MWt/MWe	1000/380	3570/1500	1190/500	392.2/150	1375/500	790/300	1600/640
- Primary Loop Arrangement	Pool	Loop/2	Loop/2	Pool	Pool	Pool	NA
- Primary Inlet/outlet temperature, C	371/510	395/550-530	395/550-530	386/530	NA	360/510	NA
- Primary Sodium Flow Rate, kg/sec	5,666.	9083/Loop	3028/Loop	4286	NA	NA	NA
- Intermediate Inlet/Outlet Temp, C	321/496	335/520	335/520	340/511	NA	NA	NA
- Intermediate Flow Rate, kg/sec	4.493.	7583/Loop	5056/Loop	3607.2	NA	NA	NA
- Number of IHX/SG Units	2/1	2/2	2/1	2/2	NA	NA	NA
- Number of Primary Pumps	4	2	2	2	NA	NA	NA
- Fuel	Metal/MOX	MOX/Metal	MOX/Metal	Metal	Metal	Metal	Metal
- Conversion Ratio	0.8 to1.3	1.1-1.3	more than 1.2	NA	< 1	NA	1.0
- Cycle Length, mo.	23.	17	17	18	40 years	NA	NA
- Decay Heat Removal	Fully Passive RVACS	Two IRACS + one DRACS	2PRACS + One IRACS	Fully Passive RVACS	Fully Passive RVACS	DRACS	NA

Table B-2: Fast reactors-world wide.

NAME	LOCATION	PURPOSE	OPERA-TIONAL	SHUT-DOWN	POWER (MWt)	POWER (MWe)	FUEL	COOLANT
France								
Rapsodie	Cadarache	Test	1967	--	40	--	U02/Pu02	Na
Phenix	Marcoule	Prototype	1974	--	560	250	U02/Pu02	Na
SuperPhenix	Creys Malville	Demonstration	1985	--	3000	1240	U02/Pu02	Na
INDIA								
FBTR	Kalpakkam	Test	--	--	42.5	12.4	(Pu+U)C	Na
ITALY								
PEC	Brasimone	Test	1981	--	120	--	U02/Pu02	Na
JAPAN								
Joyo	Oaral	Test	1978	--	100	--	U02/Pu02	Na
Monju	Ibarakl	prototype	1993	--	714	300	U02/Pu02	Na
UK								
DFR	Dounreay	Test	1963	1977	72	15	U-Mo	NaK
PFR	Dounreay	Prototype	1976	--	600	270	U02/Pu02	Na
USA								
Clemetine	Los Alamos	Research	1946	1953	0.025	--	Pu	Hg
EBR-1	Idaho	Research	1951	1963	1	0 2	Pu	Nak
Lampre	Los Alamos	Research	1959	1964	1	--	Pu	Na
EBR-2	Idaho	Test	1964	--	62.5	20	U	Na
Enrico Fermi	Michigan	Test	1965	1972	200	61	U-Mo	Na
SEFOR	Arkansas	Test	1969	1972	20	--	U02/PuO2	Na
FFTF	Richland	Test	1980	--	400	--	U02/PuO2	Na
Clinch River	Oak Ridge	Prototype	--	--	975	380	U02/PuO2	Na
USSR								
BR-2	Obninsk	Research	1956	--	0.1	--	Pu	Hg
BR-5	Obninsk	Test	1959	--	5	--	Pu	Na
BOR-60	Melekess	Test	1969	--	60	12	U02	Na
BN-350	Shevchenko	Prototype	1973	--	1000	150	U02/Pu02	Na
BN-600	Beloyarsk	Prototype	1980	--	1470	600	U02/Pu02	Na
BN-800	--	Demonstration	--	--	2100	800	U02/Pu02	Na
BN-1600	--	demonstration	--	--	4200	1600	U02/Pu02	Na
W. Germany								
KNK	Karlruhe	Test	1972	--	58	21	U02/Pu02	Na
SNR-300	Kalkar	Prototype	--	--	730	327	U02/Pu02	Na
SNR-2	Kalkar	demonstration	--	--	3420	1460	U02/Pu02	Na

Appendix C

Concept Group C

Medium Pb- or Pb-Bi-Cooled Systems

Appendix C

Preliminary Evaluation of Group C Concepts

Medium Pb- or Pb-Bi-Cooled Systems

Kune Suh, Ning Li and John M. Tuohy

C.1 Introduction

Nine concepts were considered within this grouping. They all share the common characteristics in that they are cooled by Pb or Pb-Bi eutectic. In addition, they were judged to be medium size with respect to their power output, with one exception. BREST-300 (M16) is being developed as a demonstration reactor toward a 1200-MWe commercial unit. The others are limited in size either through physical constraints and optimization (such as passive heat rejection condition) or by choice to be modular.

Inherently present in the Group C concepts is the potential to breed, to transmute waste radionuclides and mitigate waste disposal issues, to provide passive cooling, and to achieve higher thermal efficiencies.

Some approaches suggested new innovations beyond use of lead coolant. Concept M14 proposed the use of a supercritical steam cycle. Others, such as concept M18, proposed a particularly innovative approach to steam generation through direct contact of lead coolant and water, which has been studied partially by Russian researchers. Concept M27 proposed pebble fuel in a unique pool-type fast reactor.

Finally, concept M26, which only addressed a safe way to put steam generators inside the primary vessel, is acknowledged but will not be dealt with further in this round.

C.2 Concept Description and Comparison

C.2.1 Concept M14, Power Conversion System for Pb Alloy Fast Reactor Using Supercritical Steam

This concept seeks to utilize the high temperature capability of Pb-cooled reactors by employing a supercritical steam cycle to drive an electrical generator. Three reasons are given for pursuing this concept:

- To prevent lead freezing during normal and upset conditions
- To eliminate fatigue cracking of steam generator tubes in the transition zone from boiling
- To achieve high thermal efficiency.

C.2.2 Concept M16, BREST (300 MWe)

The BREST reactor and fuel cycle system concept has been under study in Russia for a decade. It is a lead-cooled fast spectrum reactor. Designers envision two sizes of BREST reactors, a 300-MWe mid-size reactor (considered here) amenable to use as a test or demonstration plant, and a 1200-MWe version, which is viewed as the best candidate for large-scale deployment. Neither design incorporates Western-style reactor containments, consistent with prior Russian liquid metal-cooled reactor design practice.

BREST reactors have the pool-type primary system arrangement, wherein the reactor core and other major primary-coolant system components (pumps, steam generators) are contained in a single, contiguous vessel.

The BREST reactor is designed to operate in a mode of slight net breeding, and thus capture the resource utilization benefits of breeders, and simultaneously utilize the large stocks of plutonium (via the initial core loading) that are accumulating from operation of thermal reactors as well as from release of excess plutonium from weapons programs.

Beyond the efficient use of uranium resources, other important goals for the BREST system include:

- Minimizing environmental impacts of waste
- Enhanced proliferation-resistance
- “Deterministic” (i.e., passive) safety
- Competitive economics.

C.2.3 Concept M18, Lead-Bismuth-Cooled Fast Reactor with In-Vessel Direct Contact Steam Generation for Actinide Burning and Power Production

This is an innovative fast reactor concept that eliminates the need for steam generators and main coolant pumps and thus offers potential for substantial capital and operating cost reduction.

The primary coolant is lead-bismuth eutectic, which flows through the core and removes the heat generated by fission in the fuel. Slightly subcooled water is injected into the hot primary coolant pool above the core. The direct contact heat transfer between the fluids causes water to rapidly vaporize, leading to the formation of steam bubbles in the reactor chimney. The large density difference between the chimney and the down-comer provides the pressure head that drives the natural circulation of Pb-Bi in the vessel. The reactor “chimney” is partitioned into square channels (one per fuel assembly) to prevent radial drifting of the steam bubbles and to maintain a uniform steam distribution. At the pool free surface, steam and Pb-Bi are separated by gravity and most residual liquid-metal aerosols are collected in the steam dryer. Then the steam is sent to the turbine and operates a Rankine cycle analogous to a BWR that can achieve thermal efficiencies above 32%. However, contrary to BWRs, this design offers the possibility to superheat the steam and achieve even slightly higher thermal efficiencies.

C.2.4 Concept M19, Lead or Lead-Bismuth Cooled Fast Reactor that Produces Low Cost Electricity and Burns Actinides from LWR Spent Fuel

The proposed reactor concept is a pool-type fast reactor cooled by lead-bismuth eutectic, that can carry out a variety of missions, including high-efficiency, low-cost electricity production, LWR spent-fuel actinide burning, or breeding plutonium (if needed). The annulus between the core barrel and the reactor vessel is used to accommodate the steam generators and the circulation pumps

Circulation of the primary coolant within the pool is based on a dual-free-level approach. The primary coolant flows from the core outlet to the hot “free level” From there it flows through the steam generator and surfaces again at the cold “free level,” from which it is pumped down to the core inlet. This circulation scheme prevents entrainment of steam bubbles to the core in the case of a steam-generator-tube-rupture event, which could otherwise cause undesirable reactivity perturbations. The dual-free-level approach can also be adopted, with minimal pressure losses, if the reactor is to operate in the natural circulation mode (i.e., no circulation pumps).

The main reactor vessel has a 6-m outside diameter, 19-m height and 10-cm thickness. A guard vessel surrounds the main reactor vessel. The guard vessel is welded to the liner of the reactor compartment. Both the guard vessel and the liner are sealed tightly, thus effectively forming the reactor containment. In a hypothetical loss-of-primary-heat-sink event, the residual heat is removed by an RVACS-type passive decay heat removal system, i.e., the decay heat is discharged through the vessel and the guard vessel to air in natural circulation on the outer surface of the guard vessel. The gap between the vessel and the guard vessel is filled with liquid lead-bismuth to enhance heat transfer. Also, a perforated cylinder is placed around the guard vessel to increase the heat transfer surface to the air. The RVACS is a completely passive system and, for the reference dimensions of the vessel, it can safely remove the decay heat from a 1,000-MWth core without violating the temperature limits of the fuel, cladding, and vessel. Therefore, the nominal reactor power is selected to be 1,000 MWth.

C.2.5 Concept M21, Integral Lead Reactor (ILR) Concept

This concept seeks to establish an “electric power generation center” that contains reactors for power generation, fuel fabrication facilities for fabricating the fuel that feeds the reactors, and fuel cycle facilities to reclaim the fissile and fertile material in spent fuel and to condition the waste products for disposal.

A lead-cooled reactor charged with metallic or nitride (U-TRU) fuel is proposed. A pool-type or top-entry type primary circuit with integrated steam generator is being considered. The core is envisioned to contain hexagonal fuel assemblies with three types of control assemblies: primary, secondary, and “ultimate safety.” The reactor is proposed to be fabricated in modules of ~350 MWe. Lead freezing in operation is mitigated by use of heat source assemblies charged with “problematic fission products.”

C.2.6 Concept M23, Lead- or Lead-Bismuth-Cooled Fast Reactor for Minor Actinide Burning

The primary goal of the proposed reactor concept is significant reduction in repository actinide radiotoxicity at lower cost than the accelerator-driven systems while maintaining acceptable safety parameters. A synergistic combination of PWR and Minor Actinide Burner Reactor (MABR) systems is proposed to achieve this goal. This combination burns plutonium (85–90% of LWR TRU) in the PWR fleet and minimizes the required rating of the latter MABR fleet.

The MABR will be based on a core designed to be compatible with generic Pb-Bi cooled fast reactor plant technology, because a hard spectrum maximizes the burning (fission) to transmutation (capture) ratio of the actinides. A major emphasis will be placed on controllability because of the small delayed neutron yield and Doppler reactivity feedback of MABR cores.

The reactor system proposed is identical to that of Concept M19, which is a pool-type fast reactor cooled by a lead alloy. See the M19 description.

C.2.7 Concept M27, Lead- or Lead-Bismuth-Cooled Fast Reactor with Pebble Fuel that Produces Low-Cost Electricity and Burns Actinides from LWR Spent Fuel

The proposed reactor is a pool-type fast reactor cooled by lead-bismuth eutectic and loaded with pebble fuel that can carry out a variety of missions, including high-efficiency low-cost electricity production, LWR spent-fuel actinide burning, or breeding plutonium.

Circulation of the primary coolant within the pool is based on a dual-free-level approach, as proposed for Concept M19. However, in this case the primary coolant flows down through the pebble bed and from the core bottom up to the pump and to the hot free level. From there it flows through the steam generator and to the cold free level, before heading down to the core inlet. This circulation scheme prevents dragging of steam bubbles to the core in the case of a steam-generator-tube-rupture event, which could otherwise cause undesirable reactivity perturbations.

The core consists of a bed of fuel pebbles contained in a large perforated bucket. The pebbles are kept still in a critical cylindrical configuration (at the bottom of the bucket) by the downward flow of the primary coolant. When the flow is interrupted because of a pump trip or a loss-of-onsite-power event, the pebbles, which are less dense than the coolant, naturally float out of the bucket and into a subcritical annular configuration. The decay heat is then removed radially by conduction across the pebbles, the vessel and the guard vessel and discharged by convection and radiation to atmospheric air at the outer surface of the guard vessel. The nominal reactor power is selected to be 1,000 MWth.

C.2.8 Concept M29, RBEC Lead-Bismuth-Cooled Fast Breeder Reactor as part of a Multi-Component Nuclear Power Structure

This concept is similar in nature to the comprehensive approach contained in the concept M21 proposal for an electric power generation center. In this case a center comprised of three types of power generators is envisioned wherein 40% of the power is produced by fast breeder reactors which serve as the source of fissile material for the complex. About 50% of the power is expected to be produced by thermal reactors that will serve to generate electricity or industrial heat while consuming fissile material. Finally, the balance of the three-part complex is a critical or accelerator driven subcritical assembly designed to alter and thereby minimize the amount of long-lived wastes, while at the same time generating radioisotopes for commercial use. RBEC may be the only concept submitted to Gen IV that is explicitly a symbiotic system involving thermal and fast reactors, and accelerator-driven transmuter systems.

This concept proposes to build on 40 years of Russian experience to provide a working system using traditional technology that is claimed to be proven (i.e., submarine experience), with an integral layout of the primary system with hexagonal fuel assemblies, high-density MOX fuel. Proponents also propose to investigate more advanced concepts using pure lead coolant and square fuel assemblies.

C.3 Potential for Concept Meeting the Generation IV Goals

C.3.1 Evaluation against Criteria/Metrics

The BREST concept is the most mature and complete of those contained within this group. It was used as the primary basis for many of the assessments made below. Each of the concepts share a common coolant type (Pb or Pb-Bi) and consequently share in a common set of benefits and development issues.

Overall, one may conclude that if the development goals were met, the BREST systems would meet the goals established for Gen IV. However, much development would be needed. The technology development issues are the subject of the next section.

C.3.1.1 Sustainability

Goal SU1 (sustainable energy generation…): Each of the concepts rely upon a coolant that inherently promotes conditions favorable for breeding and recycle of minor actinides. Essentially complete use of the uranium resource would be accomplished.

Goal SU2 (minimize waste, reduce long-term stewardship…): The development goals for the BREST fuel cycle are aggressive (<0.1% actinide carryover to waste; Sr and Cs 95-99% extracted; I and Tc 90-99% extracted; 90–99% of Np and Cm extracted), but if met would provide major new waste disposal benefits. Russian proponents refer to their approach as “radiation-equivalent waste disposal,” claiming that the radiation hazard of the waste is equivalent to that of the uranium ore originally removed from the earth, after some period of time, perhaps 200 years. These goals are transferable to each Pb- or Pb-Bi cooled system.

Goal SU3 (…least desirable route to weapons-useable materials….): The BREST reactors are intended to operate without blankets, with a core breeding ratio only slightly in excess of unity, but sufficient to account for reactivity effects, fuel cycle logistics and uncertainties. Initial stocks of plutonium would come from processing LWR spent fuel or excess weapons plutonium under special arrangements. The fuel cycle goal is that uranium, plutonium, and minor actinides would never be separated and would be inseparable in the process. Moreover, carryover of fission products is envisioned as sufficient to ensure self-protection throughout. Aside from the initial core loading, there would be no need for transport of fissile material in this system. If development goals are met, these systems are capable of high proliferation resistance.

C.3.1.2 Safety and Reliability

Goals SR1 and SR2 (…excel in safety and reliability… very low likelihood of core damage…): The goal is for “deterministic safety,” through reliance on passive features. The large inventory of lead, the large margins to boiling, the natural circulation characteristics, the low excess reactivity, and the inherent feedbacks are all beneficial features for passive safety. Much more study is needed.

Goal SR3 (…eliminate need for off site emergency response…): These concepts have the potential for highly robust mitigation features. But degree of damage, and transport of radionuclides in worst-case scenarios, is not adequately known and requires further investigation. Eliminating containment as some concepts suggest may not be practical.

C.3.1.3 Economics

Goal EC 1&2 (…clear life-cycle cost advantage, comparable risk…): The stated BREST cost goal is to be competitive with, or improve upon, LWR costs. The strategy is to take advantage of the “natural safety” characteristics to simplify the design, reduce construction commodities, reduce requirements on equipment performance (hence, we conjecture, the severity of the codes and standards adopted), and to reduce the number of construction and operating personnel.

The BREST design as well as the other Pb and Pb-Bi concepts take advantage of the inherent characteristics of the coolant to eliminate certain safety-related systems. Arguments can be made to eliminate the containment system, as well as the intermediate coolant loop. Decay heat rejection diversity and redundancy is reduced, and reactivity control system performance requirements are relaxed (at least when the core breeding ratio is near unity). Inert gas blanketing associated with sodium-cooled systems is unnecessary. In addition, regulatory influences will have to be factored into the analysis as a major uncertainty influencing the final design and ultimate cost. At the same time, existing regulatory convention developed mostly for the sodium-cooled fast reactors may need changes to accommodate the different design and deployment principles.

C.4 Technical Uncertainties

Three major technical areas (common to all lead-alloy reactor concepts) are identified that are in need of further extensive research and development before these reactors can be deployed safely, reliably, and economically. These areas are (1) neutronics core design, (2) fuel performance, and (3) compatibility of the structural materials with the coolant. Two additional issues specific to concept M18 would need to be addressed in the future, as well. These are (4) liquid metal embrittlement of the turbine components caused by lead-bismuth aerosol carried by the steam and (5) radioactive polonium transport by the steam to the power conversion cycle.

4 For those systems with high minor actinide loadings, a major emphasis must be placed on core controllability because of the relatively small delayed neutron yield and Doppler reactivity feedback. Calculations show that these important safety parameters are compromised in a fertile-free core; hence, the addition of fertile material is necessary.

5 Economic viability of these reactor concepts strongly depends on the ability of the fuel to perform satisfactorily to high burnup. However, the most promising fuel forms identified so far are either entirely new to the U.S. nuclear industry (e.g., nitride fuels) or represent a significant extrapolation of technology already developed (e.g., thorium-based metallic fuel). In both cases, it is recognized that the need exists for better knowledge and understanding of the basic properties of the fuels prior to and during irradiation (e.g., phase diagrams, thermal conductivity, diffusion coefficients, swelling characteristics, fission gas release rates, restructuring characteristics, etc.).

6 Use of heavy metal coolants in a power-producing reactor strongly depends on the corrosion resistance of the structural materials, in particular the fuel cladding. Currently in the US there is relatively little activity in this field. If an accelerated deployment schedule is to be pursued for any heavy-liquid-metal reactor concept, the activity in the cladding and structural materials area should be expanded/organized to identify suitable materials for the fuel, fuel cladding and the core internals. Also, operating envelopes for these materials need to be generated as a function of coolant type, temperature, fast fluence, burnup and oxygen concentration.

7 Separation of Pb-Bi and steam in the steam dryer of concept M18 is not complete. A small amount of Pb-Bi aerosol remains entrained in the steam stream and is carried over to the turbine, which may cause liquid metal embrittlement of the stressed parts of the turbine (e.g., the blades and casing).

8 Direct contact of lead-bismuth and steam significantly aggravates the issue of coolant activation. The primary and secondary coolants (lead-bismuth and water, respectively) are not physically segregated, and a substantial amount of radioactive polonium (the main product of bismuth neutron activation) may be released into the secondary system. This may make access and maintenance of the power cycle components costly. The concentration of polonium in the primary coolant (and thus release of polonium to the steam) can be reduced significantly by making use of an online polonium extraction system. Some potentially effective polonium extraction techniques have been identified, but they need extensive R&D. These include formation and stripping of polonium hydride, high-temperature evaporation of the lead polonide, extraction of sodium polonide from a bath of molten lead-bismuth and sodium hydroxide, and formation and filtering of rare-earth polonides.

All of the Group C concepts involve a closed fuel cycle. Yet the fuel cycle technology is generally mentioned only in passing. The pyroprocess is stated as the technology of choice for the several concepts employing Th-U-Pu-MA-Zr metal fuel, yet no electrochemical process flow sheet has been proposed for such a fuel choice. For the nitride-fueled concepts, the fate of N-15 must be specified if the initial fuel is to be enriched in N-15.

C.5 Overall Concept Potential

The Pb- and Pb-Bi-cooled fast reactors with an actinide burning core have the potential to meet all Generation IV goals. TWG 3 accepts them for further evaluation.

References

1. Concept M14: Power Conversion System for Lead-Alloy Fast Reactor Using Supercritical Steam, Dr. G. Tsiklauri et al. (PNNL).

2. Concept M16: BREST System, Dr. Michael J. Lineberry.

3. Concept M18: Lead-Bismuth-Cooled Fast Reactor with In-Vessel Direct-Contact Steam Generation for Actinide Burning and Power Production, Jacopo Buongiorno (INEEL).

4. Concept M19: Lead- or Lead-Bismuth-Cooled Fast Reactor that Produces Low-Cost Electricity and Burns Actinides from LWR Spent Fuel, Philip E. MacDonald (INEEL)

5. Concept M21: Integral Lead Reactor Concept (ILR), Jamil Alves do Nascimento et al. CTA/IEAv.

6. Concept M23: Lead- or Lead-Bismuth-Cooled Fast Reactor for Minor Actinide Burning, Neil Todreas (MIT).

7. Concept M27: Lead- or Lead-Bismuth-Cooled Fast Reactor with Pebble fuel that Produces Low-Cost Electricity and Burns Actinides from LWR Spent Fuel, Philip E. MacDonald (INEEL).

8. Concept M29: RBEC Lead-Bismuth-Cooled Fast Breeder Reactor as Part of a Multicomponent Power Structure, Pavel Alekseev (Kurchatov Institute).

Appendix D

Preliminary Evaluation of Group D Concepts

Small Pb- and Pb-Bi-Cooled Systems

Appendix D

Preliminary Evaluation of Group D Concepts

Small Pb- and Pb/Bi-Cooled Systems

David C. Wade and Orlando J. A. Gonçalves Filho

D.1. Introduction

The concepts in group D (listed below) focus primarily on developing countries that are expected to have a heavy energy demand during the next decades but have limited financial resources and infrastructure (including enough qualified personnel) to install and run nuclear power stations and fuel cycle facilities. In this context, features such as small size, modularity, factory fabrication, long core lifetimes, overland transportability, and short construction schedules are exploited to overcome the disadvantages of a lack of economy of scale. The intensive use of passive safety features simplifies their design and operation and maintenance costs, further improving their commercial competitiveness. Their modular characteristics, in particular, by allowing increments in the plant power electrical capacity, also make them attractive for local deployment in industrialized countries.

All concepts in this group, except M24 (4S), use lead or lead/bismuth alloy as the primary reactor coolant to eliminate the drawbacks associated with a positive void coefficient and a highly water-reactive sodium coolant. Use of heavy metal coolants may offer an additional capital cost advantage for these concepts, through the elimination of the intermediate heat exchanger, thereby improving their competitiveness potential.

M24 (4S), a sodium-cooled reactor concept for near-term deployment, has been included in group D for two reasons: firstly, it is a reactor of small power rating (50 MWe) designed to fulfill the same distributed market function as the other lead or lead-bismuth coolant concepts in the group, and secondly, its sponsors are currently pursuing further improvements on reactor performance (regarding economics and safety) with various new design approaches meant for the 2030's market. One prominent approach is to eliminate the intermediate heat exchanger system by changing the coolant to lead/bismuth eutectic, utilizing the technology basis developed by the study on the encapsulated nuclear heat source concept (M11), also included in this group.

M17 (STAR-H2) is a unique concept in group D. It proposes to expand the role for nuclear energy by using the reactor heat source to drive a thermo-chemical water-cracking plant for hydrogen production for the nonelectrical market of the mid to late 21st century. A desalinization bottoming cycle can also be integrated into the system for production of potable water.

In this report, an effort has been made to identify the design features, performance characteristics, and R&D needs *common* to all concepts, and those that are *unique*, hoping that this approach will make easier the decision concerning the application of available R&D funds.

D.2. Concept Group Description *(concept list; basic design features and performance characteristics)*

- Concept list (refer to Table 1 in the Introduction Section, for more detail)

Number	Concept Name	Sponsorship
M 11	Encapsulated Nuclear Heat Source (ENHS)	UCB & LLNL
M 13	Secure Transportable Autonomous Reactor (STAR-LM)	ANL
M 17	Secure Transportable Autonomous Reactor (STAR-H2)	ANL
M 24	Super Safe, Small, Simple LM Reactor (4S)	CRIEPI

- *COMMON* Design Features and Performance Characteristics

The major *common* design features and performance characteristics of this group of reactors are listed below, followed by a list of *unique* features and characteristics identified per concept (they are referred to in *qualitative* rather than in *quantitative* terms):

- Small, modular-size, pool type reactors with:
- Low power output—ranging from 125 to 400 MWth
- Low power density
- Long refueling intervals—15 to 30 years
- Lead or lead/bismuth coolants (except M24: sodium)
- Metallic or nitride fuel—U, Pu, MA, LLFP, no blanket
- Fuel cartridge/cassette factory-fabricated and overland transportable (including reactor internals, intermediate heat exchanger (IHX) and electromagnetic pump (EMP), for M24)
- No on-site fuel cycle facilities (regionalized fuel cycle services)
- Natural convection flow for primary heat transport (except M24: IHX and EMP)
- Autonomous following of generator load variations for a wide range of nominal power
- Elimination by design of severe accidents scenarios leading to core damage
- Passive decay heat removal and passive containment vessel cooling
- Simple design (no pumps and intermediate heat exchanger; except M24)
- No on-site refueling mechanism
- No mechanical connection among the fuel cartridge/module and the steam generators
- Simple reactivity control systems (movable reflector mechanism);
- No safety function for balance of the plant; simple O&M leading to reduced costs through staffing reductions

- *UNIQUE* design features and performance characteristics

<u>ENHS</u> (M11)
- A novel intermediate heat exchanger integrated within the vessel wall
- Injection of the cover gas into the coolant to increase the head for the coolant circulation allowing for the reduction of the height and mass of the fuel module (alternative design)

<u>STAR—H2</u> (M17)
- Production of hydrogen through a thermo-chemical water-cracking plant (electricity and potable water; optional products)
- High temperature coolant service conditions
- High reactor heat source at core outlet temperature
- Lead-helium intermediate heat exchanger

4S (M24)

- Annular IHX and EMP
- No rotating mechanism of the shield plug
- Sealed concept of the primary cover gas system (no piping or valves to be inspected during operation).

D.3. Potential of the Concept for Meeting the Generation IV Goals

- Sustainability

For the concepts in group D, the long core lifetime and the multiple recycling of the spent fuel reduce considerably the depletion of nuclear fuel resources and the accumulation of high level waste in comparison with the once-through fuel cycle presently used in LWR systems. The small size of the reactors and no on-site fuel cycle facilities leads to a small disruption of the natural system in the *host* countries. The environmental impact in the supplier countries is difficult to assess at this stage due to the technical uncertainties and present limited specification of fuel reprocessing technologies.

The reactor concepts in this group generally incorporate a combination of technological and material barriers that offer improved proliferation resistance. The use of the regional fuel cycle facilities can reduce the dispersion of such facilities while permitting the widespread use of nuclear power plants.

- Safety and reliability

The reactor concepts in group D make intensive use of passive safety systems for the reactor shutdown, decay heat removal and for cooling of the containment vessel in case of postulated accidents. The reduced maintenance and inspection requirements for the reactor components results generally in high reliability of the plants.

The small core sizes reduce the positive coolant void worth, and the use of lead or lead-bismuth as coolant eliminates the drawbacks generally associated with sodium-water reaction. For the sodium-cooled M24 reactor concept, the void reactivity and all reactivity temperature coefficients are negative.

Use of lead or lead-bismuth coolants, however, raises some safety and reliability concerns, such as the corrosion of the structural materials; the production of volatile and radioactive Po-210; the increased load on the reactor supporting structures, which are aggravated by the high service temperatures (>500 C) and long core lifetime, particularly for the M17 (STAR-H2) concept. A major research and testing program on the coolant/fuel/clad system and on components has to be established to address these and related problems.

- Economics

The concepts in group D incorporate a series of design policies (elimination of intermediate heat exchanger; modular construction; factory fabrication; no on-site refueling, for instance) and design features (such as passive decay heat systems, movable reflector mechanism, no control rods) that contribute to the reduction of capital, financial, operational and maintenance costs. Whether these features will overcome or not the penalty incurred by derating the core to a low power density in order to reduce pumping power requirements and to reach a long core lifetime has yet to be demonstrated.

D.4. Technical Uncertainties (*status of the technology, R&D needs)*

- Status of the Technology

In spite of Russian experience with lead-bismuth cooled nuclear submarine reactors, and the apparent disadvantages of sodium, lead and lead-bismuth alloy technologies have yet to be mastered. It should be highlighted that the experience of using sodium coolant has been gained under conditions of industrial operation of nuclear power plants, whereas Russian experience with lead/bismuth requires adaptation to new scale conditions. Lead or lead-bismuth alloy has the advantage of a higher boiling point, slight chemical activity with water, and a coolant void reactivity that is much less positive than that of sodium, or even negative. As noted above, however, there are significant disadvantages, such as high melting temperature (lead), corrosion of structural steels, pumping power requirements that dictate low power density and low coolant velocity, and production of highly volatile polonium Po-210, which would require an extensive development and testing program before commercial exploitation.

For the sodium-cooled M24 concept, many of the plant technologies have already been developed or are currently under development.

For details on fuel, coolant, and fuel cycle, see the base technology evaluations (Appendix E).

- R&D needs

Major *Common* technological needs

Coolant

- Minimization of the structural materials corrosion by Pb and Pb/Bi (chemistry monitoring and control of raw materials)
- Trapping or removing corrosion products/impurities from the Pb and Pb/Bi coolants
- Assessment of the potential for degradation of cooling capacity by sludge buildup in the event of loss of control of coolant chemistry and/or by coolant solidification in the event of local system cooldown
- Handling of the Po-210 produced in the coolant (lead-bismuth)

Materials

- Selection of corrosion-resistant structural materials
- Development of even more resistant structural materials that may be used at still higher temperatures which at the same time lead to low fabrication costs

Safety

- Demonstration of 100% natural circulation of the primary and secondary coolants
- Demonstration of reactor startup without forced circulation
- Demonstration of reactor ability to operate autonomously to compensate generator load variations
- Development and demonstration of reflector drive mechanisms
- Demonstration of reactor ability to withstand the worst conceivable accidents without damage to its fuel and structure
- Demonstration of the passive shutdown systems, passive decay heat removal and passive containment cooling
- Demonstration of the feasibility of installing and removing the fueled cartridge/module in and from the reactor pool

Economics

- Development and qualification of fabrication technologies for the low-cost serial factory fabrication of reactor modules and of refueling cassettes
- Demonstration of the economic viability of the reactor concept

Fuel Cycle

- Development of both the technologies and institutions for regionalized fuel cycle service.

- *Unique* technological needs

  ENHS (M11)

  - Demonstration of the proper functioning of the IHX rectangular channels
  - Demonstration of the proper functioning of the steam generator (efficiency of the secondary system)
  - Demonstration of the fuel cartridge/module feasibility to serve as a spent-fuel shipping-cask

  STAR—LM (M13)

  - Demonstration that nitride fuel and potential alternative fuels such as U-Pu-Zr metal fuel can be used together with Pb/Bi eutectic coolant at the projected operating temperatures and for the anticipated core lifetimes
  - Extend HT9 ferritic steel corrosion data base for cladding materials

  STAR—H2 (M17)

  - Choice and qualification of cladding and structural materials for 780°C service conditions in lead
  - Understanding the phenomenology and consequences of nitride fuel dissociation under high-temperature accident conditions
  - Development of the thermo-chemical water cracking process beyond the bench scale already achieved in Japan
  - Completion of the development and demonstration of the H_2/O_2 combustion turbogenerator or the fuel cell, as well as the 500 MW_e H_2/O_2 combustion gas turbogenerator

  4S (M24)

  - A burnable poison characteristics test for 30-year core life and the critical test by a fully scaled model of 4S remain for verification
  - Completion of the test of a large diameter coiled EMP (approximately 2m) with a capacity of 160 m^3/min flow and 2.5 kg/cm^2 head.

D.5. Overall Concept Potential versus R&D Risk

Concepts M11, M13, M17, and M24 should be accepted for more detailed analysis and evaluation. The scoresheet appears in Section 5.2.

Appendix E

Base Technology Evaluations

Appendix E

Base Technology Evaluations

SECTION E.1

STATUS OF FUEL TECHNOLOGY FOR GENERATION IV LIQUID METAL REACTORS

I—MIXED OXIDE

J. L. Carbonnier, Y. Sagayama, and R. P. Omberg

A—Status of Oxide Technology

The development of mixed oxide fuel (PuO_2—UO_2) was a cornerstone of liquid metal reactor programs around the world for over 20 years. This development culminated with the demonstration of high-burnup mixed oxide fuel in the FFTF, PHENIX, MONJU, and PFR in the United States, France, Japan, and the United Kingdom respectively. This was preceded by mixed oxide fuel testing in EBR-II, RAPSODIE, JOYO, and DFR. Mixed oxide was selected for this extensive testing because of the excellent burnup potential of the fuel system, the relative ease of commercial fabrication, and the proven safety response by virtue of the Doppler effect.

Although the fuel development programs in Europe were initially national programs, these programs evolved into a joint European program in the 1980s. This collaboration between France, Great Britain, and Germany brought together considerable experience covering a wide range of fuels. A summary of the European experience on fast reactor fuel development is shown in Table E-1 (Brown, et al.)

Table E-1. Summary of European experience in fast reactor fuel development.

	Oxide	Metal	Nitride
Number of Pins	265,000	500	~ 30
Maximum Burnup (at %)	21.7	18.0	7.0
Burnup Achieved In	PFR	RAPSODIE	PHENIX
Maximum DPA	143	77	54
DPA Achieved In	PHENIX	DFR	PHENIX

A similar table can be developed for the Japanese (Rineiskii) and American (Baker et al.) experience and this is shown in Table E-2. As in the European case, the bulk of the irradiation experience is with oxide fuel; but even though fewer metal fuel pins have been irradiated, the maximum burnup is almost as high as that of oxide. This reflects the economic incentive for high burnup. The American experience with nitride fuel in the space power program has been incorporated into the table (Makenas).

Table E-2. Summary of Japanese and American experience in fast reactor fuel development.

Japanese Experience				
	Oxide	**Metal**	**Carbide**	**Nitride**
Number of Pins	50,000	0	0	2
Maximum Burnup (at %)	13	-	-	4
Burnup Obtained In	JOYO	-	-	JOYO
Maximum DPA	100	-	-	25
DPA Obtained In	JOYO	-	-	JOYO
American Experience				
	Oxide	**Metal**	**Carbide**	**Nitride**
Number of Pins	63,500	14,109	100	76
Maximum Burnup (at%)	24.5	20	7.9	6
Burnup Achieved In	FFTF	FFTF + EBR-II	FFTF	EBR-II
Maximum DPA	200	110	?	?
DPA Achieved In	FFTF	EBR-II	EBR-II	EBR-II

The economic incentive for a lower fuel cycle cost produced a continual improvement in mixed oxide fuel. Burnup increased each year in reactors around the world, as shown in Figure E-1. This is part of an international effort to improve performance while ensuring safety and minimizing fuel cycle cost.

In the United States, three cladding materials have been employed with mixed oxide fuel: 20% cold-worked 316 stainless steel, a modified stainless steel alloy D9 with improved swelling characteristics, and a very low swelling ferritic alloy HT9. Many fuel assemblies of each of these have been irradiated in FFTF. The effect of swelling as measured by duct elongation for each of these alloys is shown in Figure E-2 (Leggett and Walters). Similar alloys have been developed in Europe. AIM1 can be considered comparable to D9, and EM10 and EM12 can be considered comparable to HT-9. Even with these improvements, the maximum fluence still remains below the goals of some projects, such as that of the European Fast Reactor (EFR). In this case, the goal is 180 dpa or 3.6 $\times 10^{23}$ n/cm^2.

There is a similar incentive for improved cladding materials in Japan. In this case, oxide dispersion-strengthened (ODS) ferritic steels are being pursued. This is driven by the economic incentive of obtaining higher thermal efficiencies by virtue of higher coolant outlet temperatures. With coolant outlet temperatures on the order of 530 to 550^{o}C and cladding temperatures above 650^{o}C, neither HT-9M nor PNC-FMS will have sufficient strength. Progress to date has largely concentrated on the fabrication properties of unirradiated material. Results to date have shown that an acceptable recrystallized structure can be repeatedly obtained with two cold-rolling passes, and that acceptable tensile and creep-rupture properties result (Ukai, et al.). Given the absence of irradiated material properties, irradiation testing for ODS ferritic cladding materials is an important objective. Future plans involve irradiating metal samples and fuel pins in JOYO and a Russian fast reactor.

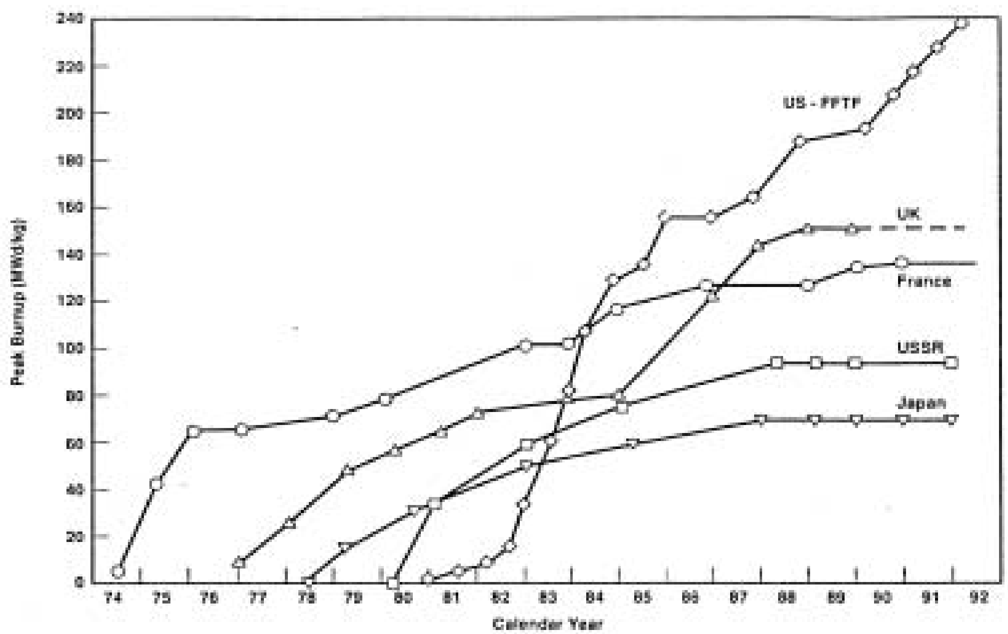

Figure E-1. Burnup of mixed oxide fuel in Liquid Metal Reactors.

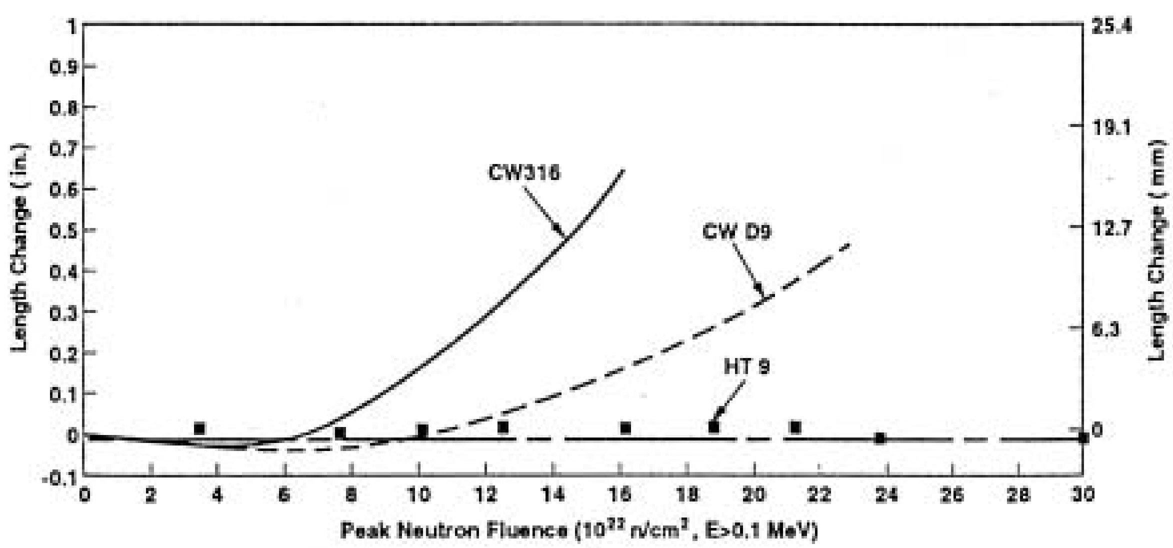

Figure E-2. Assembly elongation due to swelling in FFTF with different duct materials.

The response of mixed oxide fuel to off-normal events has been extensively examined in TREAT in the United States and in CABRI and SCARABEE in France. These tests provided data on fuel failure mechanisms, fuel motion during failure, and coolant channel blockage. This data were then used in developing and validating fuel behavior models, transient fuel performance codes, and integrated severe accident codes.

A good overall summary of transient testing experience is shown in Figure E-3. In general, the burnups are lower than the steady-state burnups shown in Figure E-1 and in Table E-1 and Table E-2. The steady-state burnups reflect the drive for better economics and fuel pins from these tests are later used in transient tests. Therefore, transient testing must necessarily lag behind steady state testing. Because of this lag, transient tests were often conducted with fuel pins that were not completely prototypic.

A summary of transient testing of advanced mixed oxide fuel, conducted in the 1980s in TREAT, is shown in Table E-3 (Wright et al.). The CDT series of tests was conducted by the Westinghouse Hanford Company to obtain transient data to support the advanced mixed oxide fuel under irradiation in FFTF. The goal residence time of this fuel was 900 full power days and the goal burnup was 150 Mwd/kg peak pellet. This was later exceeded with burnups up to 238 Mwd/kg (Baker, et al.). These tests were prototypic of advanced oxide fuel to the extent that they used mixed oxide fuel, with a larger diameter (6.858 mm) and a full-length fuel column (91 cm). But while prototypical, it should be noted that relatively few fuel pins were tested.

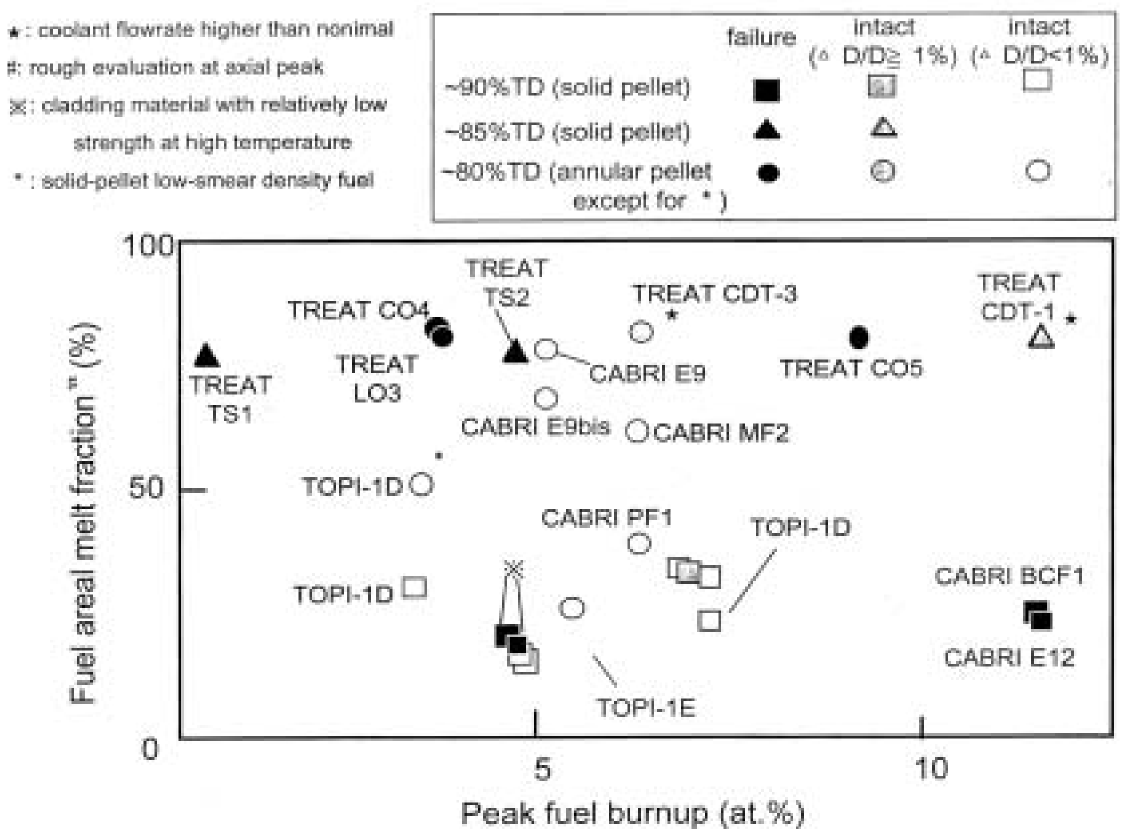

Figure E-3. Failure and nonfailure data from existing slow transient over-power tests.

Table E-3. Testing of advanced mixed oxide fuel in TREAT.

Test	Cladding	Burnup (at %)	Ramp Rate ($/sec)	Peak Power Ratio	Result
CDT-1	HT-9	12.5	0.05	4.5	Damage
CDT-2	“	11.5	1.00	16.5	Damage
“	“	6.2	1.00	16.5	Failure
“	“	6.4	1.00	16.5	Failure
CDT-3	“	6.3	0.05	4.5	Damage

B—Open Questions Related to Mixed Oxide Deployment

Given the extensive experience by several countries with mixed oxide fuel over the past 20 years, there are few technical issues that impede its deployment in sodium-cooled systems. But there is an incentive to improve economics either by increasing burnup or by increasing thermal efficiency. If the objective is to increase burnup, there is an incentive for additional data on low-swelling alloys for duct and cladding materials at a higher fluence and current operating temperatures. For increased thermal efficiency, there is an incentive for duct and cladding materials at the current peak fluence but at higher temperatures. A technological gap exists in the transient testing area. In contrast to steady state irradiation testing, only a few advanced oxide fuel pins have been subjected to transient testing. If advanced fuel were to be deployed as part of a Generation IV reactor, it would be prudent to carefully assess the extent to which the current transient testing is sufficient.

II—METAL

L. C. Walters, D. L. Porter, and R. P. Omberg

A—Status of Metal Technology

History

Metal fuel was the first fuel used in fast reactors. The simple fabrication of metal and metal alloys, the high thermal conductivity, and the relatively high fissile density all made metal fuel attractive to early reactor designers. The Experimental Breeder Reactors I and II, the Dounreay Fast Reactor, and the Enrico Fermi Atomic Power Plant all used metal fuel. The early metal fuel designs were not capable of achieving high burnup nor were they capable of performing at high sodium-coolant outlet temperatures, both contemplated in the design of future fast reactors. Therefore, development of metal fuels was discontinued in the late 1960's in favor of ceramic fuels.

However, EBR-II continued to operate with metal fuel as its main or "driver" fuel, and this reactor was the test bed for all other fast reactor fuels and materials until FFTF became operational. As a consequence, a continual development of metal fuel occurred at Argonne National Laboratory. Over a number of years, design changes were developed that increased the maximum burnup of metal fuel. And during the same period, reactor coolant outlet temperatures were generally lowered. As a result metal fuel became a viable alternative to ceramic fuel.

The concept of an Integral Fast Reactor (IFR) using metal fuel, the pyroprocess and a collocated fuel cycle facility was developed at ANL in the early 1980s. General Electric developed a similar concept using a collocated fuel cycle facility, the PRISM reactor system design. PRISM used metal fuel as the reference fuel design with mixed oxide fuel as a backup. A key aspect of these concepts was remote fabrication and electrochemical reprocessing, i.e., the pyroprocess, the goal of which was a simplified, inexpensive process, and improved proliferation resistance.

Design Characteristics

A metal fuel pin consists of a solid fuel slug that has sodium in the gap between the fuel slug and the steel cladding. Early fuel designs had little or no gap between the fuel and the cladding. When the fuel swelled due to solid and gaseous fission product accumulation, the fuel would contact and stress the cladding, leading to cladding failure. Alloying and heat treatment of the fuel to reduce swelling as well as the use of stronger cladding material did not appreciably increase the allowable burnup.

In contrast, increasing the gap between the fuel and cladding to an effective smear density of 75% or less, allowed the fuel to swell as a result of fission gas accumulation. When designed with a smear density of 75%, the porosity produced by the fission gas would interconnect just as the fuel reached the cladding. The fission gas would then be released to the gas plenum, reducing the driving force for further swelling and thereby limiting the stress on the cladding. The interconnected porosity not only allowed the gaseous fission products to escape to the gas plenum, but the porosity also allowed the fuel to flow plastically into the porosity as the solid fission products accumulated, which also limited the stress applied to the cladding by the fuel.

The solidus temperature of uranium-plutonium fast reactor fuel was unacceptably low, and so alloying elements are added to increase it. Of the many alloying elements that could be added, zirconium was chosen because the uranium-plutonium-zirconium alloys exhibited superior compatibility with the cladding materials, and because zirconium is compatible with the pyroprocess. The zirconium content of the alloy is limited to 10 weight percent; additional zirconium would require casting temperatures that would challenge the use of simple and efficient casting equipment, such as the quartz molds used in the casting of the fuel.

Performance

Metal fuel has been tested extensively in EBR-II and FFTF. Over 14,000 fuel pins of the IFR type have been irradiated in EBR-II and FFTF as shown in Table E-4 (Leggett and Walters).

Table E-4. Summary of IFR metal fuel testing in EBR-II.

Assembly	Burnup >10 at %		Burnup <10 at %		Total
	Pu	All	Pu	All	
Experiment	273	1611	329	1014	2625
Standard Core	-	-	-	11,484	11,484
Total	273	1611	329	12,498	14,109

For one period during the operation of EBR-II, the entire core was composed of either binary fuel (U-Zr) or ternary fuel (U-Pu-Zr). Individual ternary metal fuel rods have been irradiated to as high as 20 at% without failure.

Eight separate metal fuel test assemblies were irradiated to burnups greater than 10 at % in the FFTF (Pitner and Baker). These tests were intended to support the IFR and PRISM reactor designs, and were part of planning to convert the core of FFTF to metal fuel. All of these tests employed full-length fuel columns of 91.4 cm with 6.86 mm outer diameter cladding, and almost all tests used HT-9 cladding. Most of the tests were binary fuel as this was to be the reference fuel for FFTF. Two of the tests did contain ternary fuel, and so these are more prototypic of Generation IV reactors. A summary of FFTF testing is shown in Table E-5. Note that this table reflects the drive for higher burnups.

Table E-5. Metal fuel tests irradiated in FFTF.

Test	Fuel	Cladding	Peak Burnup (Mwd/kg)	Peak Fast Fluence (10^{23} n/cm^2)
IFR-1	Binary & Ternary	D-9	94	1.54
MFF-1A	Binary	HT-9	38	0.56
MFF-1	Binary	HT-9	95	1.73
MFF-2	Binary	HT-9	143	1.99
MFF-3	Binary	HT-9	138	1.92
MFF-4	Binary	HT-9	135	1.90
MFF-5	Binary	HT-9	101	1.40
MFF-6	Binary	HT-9	95	1.28
ALMR	Ternary	HT-9	141	3.3

Transient testing of metallic fuel in TREAT was resumed under the IFR program in the mid 1980s. The M series of tests was conducted using the modern metal fuel design with smear densities that allowed for greater swelling, higher burnup, and larger margin to failure (Wright et al.). The fuel for the M series of tests was irradiated in EBR-II and was ternary fuel (U-19Pu-10Zr) with D-9 cladding and a 34-cm active fuel length. The purpose of this series of tests was to support the IFR and PRISM design efforts. A summary is shown in Table E-6 (Bauer et al.).

Table E-6. Testing of advanced metal fuel.

Test	Fuel	Cladding	Burnup (at %)	Period (sec)	Peak Power Ratio	Result
M5	U-19Pu-10Zr	D-9	0.8	8	4.3	Intact
"	"	D-9	1.9	8	4.3	Intact
M6	"	D-9	1.9	8	4.4	Intact
"	"	D-9	5.3	8	4.4	Failed
M7	"	D-9	9.8	8	4.0	Failed

As can be seen from Table E-6, the burnup of the fuel in the transient tests does lag behind the steady state burnups desired for economic feasibility, and so additional transient testing may be desirable. Also, the fuel for these tests had a short active fuel length and D-9 cladding, and so it was not entirely prototypic of PRISM or IFR. This combined with the fact that relatively few pins were tested, suggests that additional transient testing is desirable.

B—Open Questions Related To Metal Deployment

Knowledge of metal fuel is sufficiently mature that a basis for design and licensing can be developed, for the alloy that was developed in the IFR program, namely, the U-Pu-Zr alloy. Since most of the metal fuel testing was performed with shorter fuel pins and binary fuel, it will be necessary to verify codes for longer fuel pins and ternary fuel. This would be part of the activity to gain an understanding of the extrapolation from EBR-II. The allowable fuel, duct, and cladding lifetime would be determined by these codes and would be verified by actual operation in a prototype plant.

With respect to fuel fabrication, the performance of the fuel is not very sensitive to minor compositional variations and thus a fuel specification, which is based loosely on composition and more stringently on fissile content/power density and reactivity, is available. Where actinide transmutation is a design objective, the performance of the fuel with high minor actinide content, and with americium in particular, should be demonstrated with further testing.

As is the case with mixed oxide fuel, relatively few prototypic fuel pins have been subject to transient testing. But it should be noted that transient testing data are available, and that the limited results support the validity of the metal fuel design.

III—NITRIDE

Y. Sagayama

A—Status of Nitride Technology

The state of development of nitride fuel is modest when compared to that of mixed oxide or metal. Nitride fuel is attractive for two reasons. It exhibits many of the same desirable characteristics as metal fuel, i.e., high heavy metal density, good thermal conductivity, and excellent compatibility with sodium. And testing of nitride fuel for space power applications did not reveal any undesirable characteristics. But the amount of testing to date is small compared to that of either oxide or metal (Leggett and Walters). The status of nitride fuel development to date is shown Table E-7 below.

Table E-7: Nitride fuel development.

Items	A. Status of technology	B. Status in concept summary papers	C. R&D issues
(1) Fuel specification	[Fuel type (fabrication experience)] -Pellet/helium bond (over 1,000) -Pellet/liquid metal bond (tens) Low “smeared density” fuel is understood as typical high burn-up fuel concept to have benign FCMI, but no firm specification.	- Vibropacking fuel, (TRU fuel with FP) [M16] - Vibropacking fuel, (TRU fuel) [M17] - (Pu enrichment: 8wt.%, TRU fuel) [M13] [Reference fuel] Metal : [M11][M18][M19][M21][M23] UO_2 : [M2]	- Same as (3)[High burn-up fuel specification]
(2) Nitrogen isotopic composition	Requirement of N15 enrichment to exclude N14, the source of C14 by [N14(n,p)C14] Conventional technologies of N15 enrichment - NITROX - Ion exchange, etc.	- It is desirable to use enriched N15 for nitride fuel. [M16]	- [N15 enrichment technology] Economical enrichment technology development Ex:Zeolite PSA method
(3) Burn-up	[Irradiation experience] - Sodium bonded fuel : ~160GWd/t (at peak) - Helium bonded fuel : ~70GWd/t (at peak) High burn-up fuel concept is indispensable to realize economical advantage. Irradiation experiments indicated significant FCMI at extended burn-up due to high fuel swelling rate and low creep rate.	- Average discharge burn-up: □6at.% (UO_2 fuel core) [M2] - Average discharge burn-up: over 10at.% [M3] - Maximum burn-up : 105GWd/tHM (Metal fuel core) [M11] - Average discharge burn-up: 72MWd/kg, Maximum burn-up: 121MWd/kg [M13] - Maximum burn-up: 15□20at.% (Metal fuel core) (This value is based on the irradiation achievements of metal fuel.) [M19] - Average discharge burn-up: 100MWd/kgHM (Metal fuel core) [M21]	- [High burn-up fuel specification] High burn-up irradiation test of various type of fuel pin to identify the high burn-up fuel concept with benign FCMI Ex. - Large gap with LM bonding - Annular fuel with gas bonding etc. - [Core material development] Core material development for high neutron dose to achieve high burn-up Ex.: High strength ferritic steels including ODS (oxide dispersion strengthened ferritic)
(4) Core safety	Possible candidate approach 1) Utilization of inherent and/or passive shutdown mechanisms 2) CDA mitigation (Nitride fuel high temperature dissociation behavior is one of the critical issues.)	- This reactor has high passive safety characteristic, that limits the fuel temperature in accidents. (No description of thermal dissociation behavior) [M3][M13][M16] - This reactor has high passive safety characteristic, that limits the fuel temperature in accidents. Some concerns of nitride fuel high temperature dissociation behavior. [M17]	- [Inherent/Passive safety] (Precise investigation is indispensable to assure the feasibility.) - [CDA mitigation] -In-pile or out-of-pile high temperature dissociation test of nitride fuel -Transient test to evaluate fuel pin failure limit

B—Open questions Related to Nitride Deployment

The status of nitride fuel is well described in the table above. To summarize:

- The fabrication experience with vibropacking is limited and needs further development
- A low-cost technology to enrich the N15 component is needed to improve the economics of the fuel cycle
- Irradiation testing is quite limited and often not well documented
- Phenomenological characteristics affecting basic fuel design such as swelling, fission gas release, fuel-cladding chemical interaction, and thermal dissociation are not well known at higher burnups.

A considerable amount of research and development will be required to bring the status of nitride fuel up to that of either metal or mixed oxide fuel. But nitride fuel does appear to have unique safety characteristics that are superior to that of either mixed oxide fuel or metal fuel (Padilla et al.). And for selected high temperature applications such as space power, nitride fuel has unique advantages and steady state irradiation testing results have been positive.

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SECTION E.2

STATUS OF COOLANT TECHNOLOGY
FOR
GENERATION IV LIQUID METAL REACTORS

I—SODIUM COOLANT

C. Boardman and M. Ichimiya

A—Status of Sodium Coolant Technology

Background

Liquid metal cooling of reactors for the purpose of achieving a fast neutron spectrum to drive the fission chain reaction has been considered from the earliest days of the nuclear era. The motivation for a fast neutron spectrum is to increase the fission to absorption ratio of neutron interaction with the transuranic elements. In that case all transuranic isotopes function as fuel while still leaving enough excess neutrons per fission for capture in U-238 sufficient to regenerate one or more transuranic atoms for each one fissioned. Thus fast neutron spectrum reactors provide for flexibility in managing transuranic inventories in a nuclear energy enterprise.

Compared with thermal reactors, the flexibility of liquid metal cooled reactors comes with both benefits and disadvantages. On the positive side, very high power densities are possible and an ambient pressure coolant system can be used even at high (~550°C) coolant temperature—allowing for high efficiency steam cycles (~40%) and no vulnerability to LOCAs. These are favorable features regarding capital cost. High burnup of fuel has been achieved (average ~130 MWd/kg)—limited by neutron fluence on the cladding rather than reactivity loss due to fission product poisoning. This is a favorable feature for operating cost.

On the negative side, however, the materials of construction for the higher temperatures, the need for refueling under inert gas (with the alkali metals), and the need for an intermediate heat transport circuit to separate the high pressure steam circuit from the low pressure primary circuit containing radioactive coolant have traditionally more than offset the capital cost advantages. Similarly, the costs of harvesting of fuel for fast reactors via recycle has exceeded the cost of virgin uranium fuel for thermal reactors, and the larger in-core working inventory of fissile leads to larger fuel carrying changes.

Key Issues

Extensive knowledge about sodium coolant has been accumulated in various liquid metal fast reactor programs around the world for decades. For example, numerous experimental fast reactor plants with sodium coolant were constructed, such as BOR- 60 (former Soviet Union), EBR-II (United States), Rapsodie (France), DFR (United Kingdom), and Joyo (Japan). Prototype plants were built, such as BN-350 (former Soviet Union), Phenix (France), PFR (U.K.), and MONJU (Japan). Finally, larger-scale demonstration plants were completed, such as BN-600 (former Soviet Union) and Superphenix (France). In these programs, many technical issues have been addressed and knowledge about sodium coolant is believed to be mature.

The remaining issues with sodium coolant that need continued development are:

(1) How to improve the coolant-related components in order to lower the plant cost?

(2) How to minimize the risk from an energetic hypothetical core disruptive accident?

(3) How to ensure that the reliability of sodium-heated steam generators will be sufficient to limit leaks to an acceptable number?

(4) How to establish in-service inspection and repair (ISI&R) technologies, in particular for in-vessel structures?

The directions taken to address these issues include the following:

(1) Innovative concepts for components have been adopted in the cost reduction efforts made in sodium cooled fast reactor designs, such as JSFR, M-JSFR, and S-PRISM. However, very small-size tests or preliminary experiments are all that support some of these components.

(2) Passive shutdown and passive decay heat removal designs are adopted in almost every concept in the Generation IV candidate concepts with sodium coolant. As for passive shut down, various kinds of mechanisms and systems have been investigated and successfully developed e.g., the self actuated shut down systems (SASS), or gas expansion modules(GEM). Passive decay heat removal by natural circulation is easy to establish in sodium cooled reactors, since the low pressure system would not lose the coolant from the core if certain design measures were undertaken such as double wall structures to guard against pipe break. The advantage of the natural circulation in sodium is to use the air as the final heat sink, although the water system would also be available as the heat sink as long as the water feed continues. The passive attribute provided in safety systems would contribute toward very low likelihood of reactor core damage.

    Moreover, in the view of the defense in depth principle, JSFR and M-JSFR adopt a re-criticality free system in order to minimize the risk of reactor core damage. In such concepts, all radioactive materials are kept inside of the reactor vessel (RV), and not discharged inside of the containment vessel (CV) with aid of the low system pressure in a sodium coolant reactor. This fact enables us to assume the source term inside of the RV. As a result, these concepts lead to multiple fission product barriers (RV, CV, fission product retention capability of the coolant) to support the proposed elimination of the need for an offsite emergency response.

(3) Concerning the chemical reactivity with water and air, the reliability of steam generators based on the operating experience, and the experience with sodium fires are important. Details are described below.

    <u>Reliability of Steam Generators Based on the Operating Experience</u>

    Since the steam generator is one of the key components of a fast reactor plant, reliability is strictly required. Various operating experience has been accumulated since the operation of steam generators of Fermi started in 1961. All of the steam generators were installed in an intermediate heat transport system of each plant. Most of the troubles were due to defects of tube-tube or tube-tubesheet welds, while hydraulic tube vibration caused leaks in a few cases. The latter cases tended to progress to larger-scale leaks. The PFR SH-2 leak event in 1987 is the most recent experience. Hydraulic vibration caused an initial leak and more than thirty tubes secondarily failed due to overheating. However, this experience revealed that a more efficient leak monitoring and protection system could avoid such a leak escalation. A remarkably long trouble-free operation was attained in EBR-II, which used straight double wall tubes. Further experience has been obtained through the operation of sodium heated steam generators such as at the SCTI facility at ETEC in the United States (70 MW SG) and at the Oarai facilities of JNC (1-MW and 50-MW SG), etc., which are R&D test facilities without reactors. Also, sodium-water reaction experiments were carried out by use of the LLTR at ETEC and SWAT in JNC to learn about measures against such events. The experience of the operations and R&D activities on the steam generators are summarized as follows:

i. Troubles mainly due to weld defects and hydraulic vibration occurred in early-designed steam generators. Some cases among them developed to a large scale of leak that caused failure of a large number of tubes and opening of rupture discs of the pressure relief system.

ii. Nevertheless, all the steam generator troubles terminated successfully without any radiological release by the operation of the protection system such as the isolation and depressurization of the water/steam system. They revealed that the troubles could not have jeopardized the reactor safety at all. For instance, the French experience with Phenix and the Russian experience with BN-350 have shown that steam generators with adequate reliability can be constructed and when steam generator leaks do occur they can be safely accommodated and repaired.

iii. The troubles mentioned above can be avoided through operating experience, design and the development of manufacturing techniques.

iv. Some experience using double wall tube steam generators was very successful. The long operation of the EBR-II steam generators without leaks indicated high reliability of the double wall tube system. The experience at EBR-II has been most encouraging as this reactor operated for more than 28 years without experiencing a single SG leak. After 18 years of operation, one of the superheaters was removed and destructively examined. The examination did not find any "life limiting phenomena" and, in fact; the compatibility of sodium with the SG was such that the original mil rights could still be seen on the tubes after 18 years of operation.

v. However, additional effort to enhance the reliability is still necessary if the intermediate heat transport system is eliminated and steam generators are installed in a primary heat transport system in future.

It should be noted that although worldwide steam generator experience has been varied, there have been some notable successes that indicate that a properly designed and constructed steam generator will be extremely reliable. In conclusion, the present status of design and manufacturing technique is reliable enough to avoid an unacceptable scale of leak events in steam generators installed in an intermediate heat transport system.

Sodium Fires Experience and Related R&D

Since fast reactors use liquid sodium at operating temperatures above its ignition temperature, leakage of sodium to the atmosphere will result in fires. The burning of sodium produces heat and smoke, the latter known as sodium aerosols. More than a hundred sodium leakages are reported in the fast reactor plants. The largest amount of sodium leakage in these plants occurred in the Superphenix plant in 1987. Approximately 20 m^3 of sodium was gradually spilled from the spent fuel storage tank. Because all of the spilled sodium was in the gap between the reactor vessel and the guard vessel, which was maintained as an inert atmosphere, the normal design practice, no fire occurred. Other sodium leakage reported in the fast reactor plants are within a 1-m^3 amount. The most recent reported is approximately 0.7-m^3 sodium leakage in the MONJU plant. The sodium leakage and resulting fire continued more than 3 hours, and all of the sodium was burned in a 4.5-m^2 area of the floor liner. No serious damage was observed. Apart from the reactor plants, a sodium fire accident occurred in the Almeria solar plant in Spain in 1986, which is well known. The accident took place during a repair procedure of a sodium valve. Because of disassembling the valve in an inadequate plant condition, approximately 14 tons of sodium was spilled and burned. A part of the room and roof was damaged by the fire. Note that none of the sodium fire experiences in the fast reactor plants resulted in severe plant damage.

Concerning the R&D, extensive studies of the sodium fire phenomena have been carried out worldwide. Large-scale experiments have been performed in such facilities as LSFF-CSTF/HEDL (U.S.), ESMERALDA/CEA (France), FAUNA/FZK (Germany), and SAPFIRE/JNC (Japan). Also, many comprehensive computer codes have been developed and validated with the experimental data. These include SOFIRE, SPRAY, NACOM, FEUMIX, PULSAR, ASSCOPS, BOX, etc. Consequently, the phenomenon of sodium fires has been well studied.

Consideration of various countermeasures, including a detection system, a mitigation system, and an extinguishing system, can minimize the risk of the fire accident to an acceptable level in the plant design.

(4) As for continuous monitoring for sodium leakage, contact-type sodium leak detectors have been generally used for sodium components, piping, vessel, and valves. Gas sampling type small leak detection systems have been applied for the reactor vessel, and the primary and secondary cooling circuits on MONJU. Currently, techniques are being studied to achieve higher sensitivity and reliability.

Nondestructive inspection techniques for the sodium coolant boundary have also been developed. Remotely operated inspection techniques have been developed for the reactor vessel. Key technologies are high-temperature and radiation resistant sensors for visual and volumetric examination, and a robotics vehicle, which can access the narrow gap between the reactor vessel and the guard vessel. These sensors and robotics devices can be used at high temperature (~200ºC) under radiation conditions. Remotely operated systems have been applied at MONJU and Superphenix.

Visual inspection techniques for the heat transfer systems have also been developed. Monorail type ITV system for PHTS piping, which can move along the piping route and observe the outer surface of piping/thermal insulation to examine any trace of sodium leakage, has been developed and applied (MONJU). Conventional inspection techniques are used in the IHTS, where human access is easy. Valves, pump bodies, and IHXs are monitored for sodium leakage by sodium leak detectors installed around these components.

A SG tubing inspection technique has also been developed. An advanced ECT (eddy current testing) technique for ferromagnetic material such as Cr-Mo steel and an insertion technique of volumetric examination probe for long and helical-shaped tubing have been developed and applied (MONJU). Ultrasonic testing inspection system for SG tubing has been developed and demonstrated (Superphenix). In addition to ISI, a plugging technique has been developed and is prepared for repair work.

In addition to sodium boundary inspection, an in-sodium inspection technique is required to realize ISI for in-vessel structures. Also, an in-sodium repairing technique is required. Some key technologies have been studied, for instance, the under-sodium viewing technique. Horizontal USV has been applied for MONJU, and Vertical USV had been applied at PFR. Because the resolution ability of these USV techniques depends on the distance between the ultrasonic sensor and the object, advanced USV techniques have been under development to achieve in-sodium ISI for the core-support structure with high-resolution. An in-sodium repair technique needs to be developed.

In conclusion, the ISI technique for the sodium boundary has been developed and successfully applied as program shown in Table E-8:

Table E-8. Typical ISI program.

ISI plan		Periodic 100%/RY
Reactor Vessel		CM, VI, ST, VO(*1)
PHTS	Piping	CM, VI
	Pump/IHX vessel	CM, VI
	Other components (valves)	CM, VI
IHTS	Piping	CM, VI
SG	Tubes	VO
	Shell	CM

CM: Continuous Monitoring for sodium leakage, ST: Surveillance Test, VI: Visual Examination, VO: Volumetric Examination.
*1: Dissimilar weld.

Unresolved Technical Issues and Discussion of the R&D Related to Sodium Coolant Deployment

The unresolved technical issues relating to various concepts in Group A and B are summarized in the column under "Unresolved technical issues" of Table E-9. The R&D needs relating to sodium coolant technologies employed in the various concepts of Group A and B are also described in the column "Discussion of the R&D" of Table E-9. To summarize:

- As challenges toward competitiveness:
  - Development of innovative components, perhaps such as integrated components (IHX + pump), and electromagnetic pumps
  - Thermo-hydraulics examination in compact components
  - Development of new structural materials that contribute to compactness
- As challenges toward safety and reliability:
  - Demonstration tests of GEM and SASS
  - Demonstration by further in-pile testing
  - Demonstration of passive heat removal in natural circulation cooling
  - Demonstration of elevated structural design code based on inelastic analysis
  - Demonstration of seismic isolation systems
  - Demonstration of ISI&R technologies for in-vessel structures.

Table E-9: Technology evaluation (sodium coolant).

	1. Status of technologies	2. Unresolved technical issue						3. Discussion of the R&D
		M4 (JSFR)	M6 (BN-600)	M22 (RNR-1500)	M30 (CPFR)	M31 (SFR)	M32 (SIPS)	
Coolant	•Passive shutdown: •GEM: Demonstrated by in-core test. •SASS: No demonstration test. •Reflector control: Design method has been investigated. •Shutdown by inherent safety feature: Demonstrated in the experimental reactor. •Re-criticality free: Various concepts has been invented •Passive decay heat removal: Main current is a forced circulation cooling. •Elevated temperature structural design: Design codes based on elastic analysis are established, and recognized to be necessary to reduce over-conservatism. •Seismic isolation: Two-dimensional system has been commercialized. •Innovative concept of component: Very small-sized tests or preliminary experiment •Examination of thermal-hydraulics: Tested only for existing systems. •ISI & repair: ISI against vessels and piping has been developed. But ISI against under Na structure is difficult. •Structural material: •Na chemical activity:	•Demonstration of SASS. •Verification of countermeasures for re-criticality free. (FAIDUS concept) •Development of advanced structural design guide. •Development of three-dimensional seismic isolation technology. •Development of integrated component (IHX + pump). •Examination of thermal-hydraulics within the compact components.	•*Confirmation of effectiveness of upper Na plenum.* •*Development of core catcher lined with sheets of molybdenum.* •*Development of RVACS.*	•Development of ISI&R technology.	•*Confirmation on effectiveness of molten fuel discharge capability through center hole in fuel pellet.* •*Development of compact IHX arrangement.*	•Development of copper bonded heat exchanger.	•Development of copper bonded heat exchanger.	•Passive shutdown: •GEM, SASS and Reflector control: Demonstration for practical use. •Inherent safety feature: Demonstration in a large scale reactor. •Re-criticality free: Demonstration by in-pile tests. •Passive decay heat removal: Demonstration of a natural circulation cooling. •Elevated temperature structural design: Improved structural design technology, such as design by inelastic analysis, is needed. •Seismic isolation: Demonstration of three-dimensional system. •Innovative concept of component: Development of •Integrated component, (IHX + pump) •Compact IHX, and •Copper bonded heat exchanger. •EMP. •Examination of thermal-hydraulics: Examination for proposed systems. •ISI&R: Development of ISI&R technologies in Na.
		M5 (M-JSFR)	M1 (S-PRISM)	M7 (KALIMER)	M8 (LMFR + recladding)	M15 (AFR-300)	M24 (4S-LMR)	
		•Development of completely passive DHR system by a natural circulation. •••,•••Same with M4	•Passive shutdown qualification. (Demonstration of SASS and GEM). •Enhancement and qualification of RVACS. •Seismic bearing qualification. •EMP qualification. •Water simulation tests. •Thermal stripping tests.	•PSDRS performance test. •High temperature creep fatigue. •Large size EMP in high temperature environment. •Flow instabilities of once through type SG.	•,•,•Same with M7.	•*Demonstration of passive shutdown capability against ATWS events by inherent safety feature.*	•Reflector controlled core technology and reflector drive technology.	

II—LEAD OR LEAD-BISMUTH COOLANT

J. Tuohy, N.Li, and C. Boardman

A—STATUS OF LEAD OR LEAD-BISMUTH COOLANT TECHNOLOGY

Background

Pb or Pb-Bi coolant shares some characteristics with sodium, which make fast reactors attractive alternatives to thermal reactors. The use of Pb-Bi eutectic as a reactor coolant is based on over 80 reactor-years of experience in the former Soviet Union and Russian Federation[1,2]—a significant but more limited and less mature experience base compared to the worldwide experience with sodium technology.

If it turns out that evolutionary improvement on sodium-cooled fast reactors will not resolve the complexity and cost issues enough to make fast reactors competitive with LWRs,[2] then modest investments in Pb or Pb-Bi coolant may pay great dividends.

Pb or Pb-Bi offers important attributes as a fast reactor coolant: it is neutronically superior to other liquid metal coolants. It is inert, and it has a very high boiling temperature and low vapor pressure. The total core coolant void coefficient is negative. These attributes offer the prospect for designing a simple, low-cost reactor system with enhanced safety features, possibly with long core life.[3]

The heavy liquid metal coolant option may enable a new synergism among the criteria of simplification, proliferation resistance, and high inherent/passive safety (Spencer, 2000), and provides concepts of "flexible energy conversion systems" that can meet the changing and diverse market needs.[4] The current studies of potential reactor systems cooled with Pb or Pb-Bi emphasize features to reach economic competitiveness including system simplifications, standardization and "mass production" of plant modules, shipment and fast assembly of the modules at the site, once-for-life reactor fueling, and siting of single or multiple standardized energy supply modules depending on local capacity needs. Fuel conversion, rather than breeding, is the approach for ultra long core lifetime, making maximum utilization of the nuclear fuel resource (Spencer 2000). Many concepts also include transmutation as a performance goal. Many of these goals can be attained with sodium-cooled fast reactors as well, although some of the simplifications made possible by Pb or Pb-Bi are perhaps more difficult with sodium coolant.

The disadvantages of Pb or Pb-Bi coolant include its very high density, high melting point (for Pb), toxicity, requirement for corrosion protection techniques, high pumping power requirement, activation (Po-210 in Pb-Bi), activation-related contribution to the mixed waste burden, and lack of practical

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1. See, for example, B. F. Gromov et al., "Use of lead-bismuth coolant in nuclear reactors and accelerator-driven systems," *Nuclear Engineering and Design,* 173 (1997) 207–217; "Heavy Liquid Metal Coolants in Nuclear Technology," *Proceedings of HLMC-98, Obninsk, Russia* (1999).

2. See, for example, E. P. Velikhov, I. S. Slesarev, E. O. Adamov, V. V. Orlov, V. I. Subbotin, and Yu. M. Cherkashov, "The Highly Safe and Economical NPP with Liquid Lead-Cooled Reactors," report (1990); V. V. Orlov et al, "Lead Cooled Fast Reactor—New nuclear technology for the future large-scale nuclear power," manuscript; B. W. Spencer, "The Rush to Heavy Liquid Metal Reactor Coolants—Gimmick or Reasoned," *Proceedings of ICONE 8, ICONE-8729* (2000).

3. H. Khalil et al., *Preliminary Assessment of the BREST Reactor Design and Fuel Cycle Concept*, ANL technical report ANL-00/22 (2000).

4. D. C. Wade and D. J. Hill, "Requirements and Potential Development Pathways for Fission Energy Supply Infrastructure of the 21st Century—A Systems Viewpoint," *Proceedings of Global'99* (1999).

experience and relevant database outside of Russia. The toxicity of Pb and some of its compounds weighs negatively on public acceptance.

However, deployment of Pb-Bi eutectic-cooled reactors in the Russian nuclear submarines indicates that many of the technical problems can be overcome with adequate design, construction, and component manufacture methods. The problem of higher pumping power needs can be mitigated to some extent with a higher coolant volume fraction, since the high-power density and high-flow requirement in the conventional fast reactors may not be paramount for some applications. The recent disclosure of Russian's extensive technology developments involving Pb-Bi eutectic coolant presents new information, especially in the areas of corrosion protection and special alloys (Gromov 1997; HLMC-98 1999). Russia's MINATOM has been developing a Pb-cooled reactor (BREST), and IPPE (Obninsk) is promoting SVBR-75/100 reactor modules based on the submarine reactor technology. There are some international R&D investments, such as in the DOE NERI program, supporting concept studies for advanced reactor systems based on Pb or Pb-Bi. Additionally, the development of accelerator-driven systems (ADS/ATW) has generated significant interest in Pb or Pb-Bi as high-power spallation targets and subcritical transmutation blanket coolant, and this has led to international development programs and test facilities.

Compared to the state of sodium-coolant technology, Pb-Bi, and especially Pb, coolant technology is much less mature. Significant development is needed before Pb or Pb-Bi can be judged on a par with sodium as a coolant for liquid metal reactors (e.g., Khalil 2000). The roadmap evaluation will take this into account. However, in the bigger picture of a large-scale nuclear power industry, coolant technology development costs are relatively small. The constraint of limited funding should be dealt with separately in R&D planning. It is important to investigate and verify the potential benefits of Pb or Pb-Bi coolant with some fundamental R&D and concept system studies. It is also important to transfer and verify the extensive Russian Pb-Bi nuclear coolant technology, and to closely monitor the Russian development efforts.

Table E-10 shows the main thermo-physical properties of sodium, lead, and lead-bismuth eutectic (LBE, 44.5 wt% Pb, and 55.5 wt% Bi). The higher melting point of Pb (327°C) makes this heavy metal coolant significantly more difficult to utilize than LBE (125°C) or Na (98°C), since it significantly raises the allowable cold leg temperature and increases the potential for freezing. Additionally, increasing the upper operating temperatures may further burden the materials in mechanical properties and corrosion resistance. The higher density may significantly increase the weight and stress level of the major components and vessels, although in fully submersed portions the gravity pull of major components is balanced more by the buoyancy of the coolant, which is achieved in many designs. Since many components and fuels will tend to float up, hence must be held down, a different design practice must be adopted.

Table E-10. Thermophysical properties of Na, Pb, and Pb-Bi.

Properties		Units	Na	Pb	**Pb-Bi** (44.5% Pb and 55.5% Bi)
Atomic Number		—	11	82	—
Atomic Mass		—	22.9	207.2	—
Melting Temperature		°C	98.	327.4	125.
Boiling Temperature		°C	883	1745	1670
Heat of Melting		kJ/Cg	114.8	24.7	38.8
Heat of Vaporization		kJ/Cg	3871	856.8	852
Density	Solid 20°C	kg/m^3	966	11,340.	10,470.
	Liquid 450°C		845	10,520.	10,150.
Heat Capacity	Solid 20°C	kJ/kgK	1.230	0.127	0.128
	Liquid 450°C		1.269	0.147	0.146
Thermal Conductivity	Solid 20°C	W/mK	130	35	12.6
	Liquid 450°C		68.8	17.1	14.2
Kinematic Viscosity @ 450°C		m^2/s	3E10-7	1.9E1-7	1.4E10-7
Prandtl Number @ 450°C		—	0.0048	0.0174	0.0147
Surface Tension @ 450°C		mN/m	163	480	392
Volume Change with Melting		%	+2.6	+3.6	+0.5
Fast Activation Cross Section		Mb	0.67	3.6	3.4
Scattering Cross Section		b	4.0	7.5	7.55
Cost (estimated)		$/pound	0.17	0.25	1.75
Chemical Reactivity			High	Moderate/Dust	Moderate/Dust

Key Issues

1. Coolant Technology Related to Heavy Metal Coolants

Unlike Na, corrosion of the fuel and reactor systems is a major concern in HLMC designs, which can only be addressed by a major materials testing program geared to develop and refine:

- Impurity control technologies
- Corrosion rate mapping as a function of temperature, coolant velocity, oxygen content, and other parameters for a spectrum of possible reactor and cladding materials (both ferritic and stainless steels)
- Degradation of mechanical properties of structural materials, hydrodynamics, and heat transfer as a function of the operating conditions
- The contact heat-transfer resistance of oxide coatings as a function of the structural material and the operating condition (temperature, velocity, and oxygen concentration in the coolant)
- Methods of reducing the effects of polonium activation.

Russian technology and alloys, if made available, might significantly shorten and reduce the development program.

The key corrosion processes associated with HLMC reactors include:

(1) Interaction of coolant with oxide films

(2) Dissolving of steel components and their chemical interaction with non-metal impurities (oxygen, hydrogen, etc.)

(3) Penetration of liquid metal into solid materials causing frontal and inter-granular corrosion

(4) Transport of structural material along the circuit.

Pb and Pb-Bi exhibit strong corrosion (and erosion-corrosion at high flow velocities) effects on unprotected structural materials, primarily due to the nonnegligible solubility of Fe, Cr, and Ni of steels. Due to the chemical inertness of Pb and Bi, however, steel surfaces can be "passivated" with a protective oxide film that is "self-healing," i.e., by controlling the thermodynamic activity of oxygen in the coolant, the oxide film can be generated, maintained, and regenerated in situ.[5] This technique has been developed and deployed in Russia for nuclear submarine reactors (HLMC-98 1999), and is being verified and extended in many international programs.

Specifically, this oxygen control technique seeks to maintain the oxygen level in Pb-Bi or Pb such that a stable and protective oxide film (mostly Fe_3O_4, with a chromium enriched sublayer) can form and be maintained, while there is no excessive oxygen to form and precipitate lead oxide slag. The required oxygen level (in the 10^{-6}–10^{-5} wt% range) can be measured by solid electrolyte sensors and maintained by:

- Adding oxygen by bubbling oxygen gas mixtures, or passing coolant through a canister with PbO balls and dissolving the required amount.
- Removing oxygen, and periodically restoring the flow system, by sparging coolant with a hydrogen gas mixture (with helium or argon). The hydrogen reacts with the oxygen, forming H_2O vapor that is removed from the system by a cover gas system.

For some special alloys (austenitics such as 316 and ferritics/matensitics such as HT-9), addition of minor alloying components (e.g., Si and Al) and proper secondary treatment may enhance the corrosion resistance to within acceptable levels. For systems using these materials, the oxygen control technique can perhaps be applied with similar burden compared to impurity control in sodium systems.

Precipitates must be kept in suspension by maintaining coolant velocity in the range of 1–3 m/s (most Russian designs use 1.8 m/s for velocity in core), and by passing a fraction of the coolant through a filter in a bypass line to remove the precipitates and solids. Some of the precipitates may be restored during periodic hydrogen sparging.

However, the oxygen technique is yet to be fully developed and validated in large reactor systems. It may be difficult in systems with predominantly natural convection flows. Additionally, materials with sufficient corrosion resistance and stable protective oxide film for long-life cores are not proven. For reactor vessels that operate at about 400°C, where corrosion is minimal, long lifetime may not be a problem.

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5. N. Li, *Active Control of Oxygen in Molten Lead-Bismuth Eutectic Systems to Prevent Steel Corrosion and Coolant Contamination*, Los Alamos National Laboratory, LA-UR-99-4696 (1999), accepted for publication *in J. Nuc. Materials*. X.Y.He, N. Li and M. Mineev, "A Kinetic Model for Corrosion and Precipitation in Non-isothermal LBE Flow Loop," *J. Nuc. Materials*, 297 (2001) 214-219.

2. Application to a Direct Contact Steam Generator in a HLMC Reactor

The steam dryer will remove most of the Pb-Bi that becomes entrained in the steam stream, which is saturated after some time to less than 10^{-3} kg/m^3 (HLMC-98 1999). The residual droplets will be transported to the turbine, which raised the following concerns:

- Potential for liquid metal embrittlement (LME) induced failure of the turbine blades and casing
- Pb or Pb-Bi erosion of the turbine surfaces
- Contamination of the turbine plant by radioactive polonium

Each of the above phenomena is discussed below.

<u>Liquid metal embrittlement induced failure</u>

Liquid Metal Embrittlement is the brittle fracture or loss of ductility, of a normally ductile metal (stressed in tension) upon contact with a liquid metal. Tensile stresses are necessary to drive the crack growth. Although smooth samples of stainless steel do not appear to be susceptible to LME, samples where micro-cracks and discontinuities in the oxide film, which are normally present, can allow LME to occur.

<u>Characteristics of LME that can be utilized to prevent embrittlement include</u>:

LME is not a corrosion, diffusion, or intergranular penetration phenomenon. It will not occur unless the liquid is in intimate contact with the solid surface and present at the tip of the propagating crack.

- A thin oxide coating is enough to prevent the liquid metal/solid contact and LME from occurring
- LME does not take place in ceramic materials, but in metallic solids
- LME may embrittle certain metals and not others
- LME may occur at any temperature.
- LME occurs in the presence of tensile stresses, only
- LME causes severe reduction of the fatigue life of the solid.

An oxide or a cermet coating could be used to protect the stainless steel casing and turbine blades from LME. However, erosion from water and Pb-Bi droplets and/or oxide scale from the steam pipes make the viability of coating the turbine blades and casing an uncertain approach at best.

<u>Pb-Bi erosion of the turbine surfaces</u>

Liquid droplet erosion is a well-known problem shared by all steam turbines. In most cases, the source of liquid water is condensation in the LP section of the turbine, since the HP turbine is supplied with dry superheated steam. However, in this case a significant amount of relatively heavy (dense) liquid Pb-Bi may be carried over by the steam. Impingement of water droplets on the moving blades causes mechanical removal of the blade material (i.e., erosion) due to the high differential velocity and impact of the droplets on the turbine components.

Numerous remedies have been used to mitigate erosion of steam turbine blades. The most successful approaches hard face the tips of the blades. Hard facing is made of a tough hard material (typically cobalt-based alloy Stellite or a titanium-based alloy). The application of erosion shields has made liquid droplet erosion a manageable problem in steam turbines.

However, the high density of Pb-Bi makes erosion a very real concern for large steam turbines, since even a small amount of erosion will remove the protective oxide layer, making the blades and turbine casing susceptible to liquid metal embrittlement-induced failures.

Contamination of the turbine plant by radioactive Polonium

Polonium has no stable isotopes. However, Po-210, Po-214, and Po-218 naturally occur in the decay chain of U-238. Po-210 has a relatively long half-life and emits alpha particles of 5.3 MeV with 100% yield. It is produced in the lead-bismuth coolant by neutron activation of Bi209. When created by neutron activation in the lead-bismuth coolant, polonium rapidly forms a stable compound with lead, known as lead-polonide. Upon contact with water, a very volatile hydride (H_2Po) is formed. Continuous on-line polonium extraction from the Pb-Bi coolant is essential to inhibit the formation of the volatile H_2Po, in order to minimize the polonium transport out of the reactor pool. However, despite the construction of Pb-Bi-cooled reactors for submarines, there is no industrially established and proven polonium extraction technology. The problem of polonium release and transport in the Pb-Bi-cooled reactor utilizing a direct-contact steam generator is a serious issue. Radiation levels in the steam, feedwater, and major turbine system components would likely exceed the EPA limits for inhalation, ingestion, and unrestricted maintenance under any practically feasible polonium extraction rate and over the whole range of operating temperatures.

3. Miscellaneous Challenges and Concerns

Bottom Mounted Control Rod Drives

Most sodium-cooled reactors utilize top-mounted control rods. That is, the control-rod mechanisms are mounted on the reactor head above the cover gas space. The control and scram assemblies can be inserted by driving the poison assemblies into the core with a drive-in mechanism, or by releasing the rods and allowing gravity to pull them into the core.

However, several of the small modular pool type Pb-Bi-cooled reactors proposed to date have utilized bottom-mounted control rod drive mechanisms, because feedwater spargers and steam chimneys located above the core prevent use of conventional top-mounded drive mechanisms. The bottom-mounted control rod drives require a leak-tight liquid metal seal around the control rod drive shaft. This type of seal has never been utilized in any liquid metal reactor.

Floating Fuel and Control Assemblies

Because heavy metal coolants are denser than the components that make up the core, heavy metal reactor design will need some different strategy for anchoring parts and prevent the fuel, the blanket, and the shutdown assemblies from floating out of the core. Some of the special design features developed for sodium-cooled reactors will not be effective. This is not necessarily detrimental, since the much larger buoyancy can reduce the component stress, and make retrieving parts easier without complete drainage.

Higher Operating Temperatures

The freezing temperature of Pb-Bi is very close to that of sodium (125 versus 98^{o}C). It can therefore be used in a reactor that operates in the same general temperature range, as illustrated in Table E-11.

Lead is much less expensive (the bismuth adds less than a few percent capital cost at today's market price) and does not produce much polonium. However its higher freezing temperature means that a lead-cooled reactor must operate at a higher cold-leg temperature than a sodium-cooled reactor to ensure that freezing is avoided. Nonetheless, lead-cooled reactors can still operate in a temperature range not much different from that of LBE-cooled reactors.

Table E-11. Operating temperature versus coolant type.

	Melting Temp. °C/°F	Refueling Temperature °C/°F	Core Inlet Temperature °C/°F	Core Outlet Temperature °C/°F	Clad Midwall Temperature °C/°F
Sodium	98/208	232/450	371/700	527/980	604/1120
Lead	327/621	420/788	540/1004	620/1148	Not specified
Lead-Bismuth	125/257	259/498	371/700	527/980	604/1120

Gas Expansion Modules (GEMS) Will Not Work

Some sodium-cooled reactor concepts accommodate unprotected loss-of-flow events by using GEMS to increase the leakage from the core on loss of forced circulation. Due to the higher density of Pb and Pb-Bi, the GEMS will not function in an HLMC reactor. Also, the higher scattering cross section of lead reduces the potential worth of GEMS.

Reactor Size Limitations

It may not be possible to scale up the size of heavy-metal-cooled reactors due to the difficulties associated with supporting heavy-lead-containing vessels while also providing adequate flexibility in piping systems. The difficulty of the task is illustrated by the unique approach that the Russian BREST-1200 designers have proposed for their concept.

Quoting from a Russian paper[6]

"A pool-type layout of the reactor and steam generators is adopted in the BREST-1200 design, where those are installed inwardly (inside a) concrete well with thermal insulation without metal vessel. Natural convection of air, circulating through pipes (120 mm pipe size), the down-comer and up-comer legs of which are placed inwardly the load bearing concrete, is used to maintain the concrete temperature within the admissible limits. Reinforced concrete body is lined inwardly with 8-10 mm thick anchored steel. Air pipes are secured to the lining from outside, and thermal insulation is anchored from inside, being made in the form of stainless steel clad thermal insulating units of 600 mm total thickness."

Many of the concepts have adopted the principle of modular design, factory fabrication, and on-site assembly. For these concepts, scale-up is neither necessary nor desired.

B—UNRESOLVED TECHNICAL ISSUES AND DISCUSSION OF THE R&D RELATED TO LEAD OR LEAD-BISMUTH COOLANT TECHNOLOGY

Summary of HLMC technology

The unresolved technical issues and related R&D needs of lead or lead-bismuth coolant are shown in Table E-12, together with the status of technologies described above. To summarize:

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6. Filin A. I., Orlov V. B., Leonov V. N., Sila-Novitskij A. G., Smirov V. S., Tsikunov V. S., "Design Features of BREST Reactors. Experimental Work to Advance the Concept of BREST Reactors. Tests and Plans," Global-99, August 29–September 3, 1999, Jackson Hole, Wyoming, USA.

1 Corrosion

- Development of a corrosion rate mapping as a function of temperature, coolant velocity, oxygen concentration and other parameter for a spectrum of possible reactor and cladding materials (both ferritic/martensitic and austenitic steels)
- Improvement and/or development of corrosion and coolant contamination mitigation techniques (oxygen sensors, oxygen control systems, filters, etc.)
- Development of new materials of enhanced corrosion resistance at higher temperature for higher thermal-electric efficiency and other energy use applications (e.g., hydrogen production)
- Investigation of mechanical property degradation of structural materials
- Investigation of thermal hydraulics (the effect of protection technique on contact heat transfer)
- Investigation of radiation damage of materials—especially if the Western alloys have to be modified for sufficient corrosion resistance

2 Design, Operation and Maintenance

- Development of refueling technique (refueling system, HLMC removal, etc.)
- Improvement of mitigation measures against Po hazard
- Development of design strategies to address floating tendency of components and control assemblies
- Development of ISI&R technologies (may borrow from sodium technology)
- Development of seismic isolation system

3 Safety and Reliability

- Demonstration of SASS
- Demonstration of natural convection cooling in prototypic test facility
- Development of re-criticality free system and demonstration for practical use

4 Innovative Concepts of Components (e.g., direct contact steam generator)

- Investigation of potential liquid metal embrittlement-induced failure of turbine blades and casing
- Investigation of Pb or Pb-Bi erosion of surfaces
- Determine level of turbine plant contamination by radioactive polonium.

Table E-12: Technology evaluation (lead-bismuth coolant).

	1. Status of technologies	2. Unresolved technical issue	3. Discussion of the R&D
Coolant	(1) Passive shutdown •SASS: Impossible use of gravity driven scram •GEM: Not Work •Shutdown by inherent safety feature: No demonstration test (2) Re-critically free: •Detail investigation is necessary (3) Passive Decay heat removal: •No demonstration test (4) Elevated temperature structural design: •No design code applicable to HLMC reactor (5) Seismic isolation: •Two-dimensional system has been commercialized. (6) Innovative concept of component: •Detail investigation is necessary (7) Examination of thermal-hydraulics: •Very small-sized tests or preliminary experiment (8) ISI&R: •Difficulties due to opacity of HLMC. (9) Material against Corrosion Problem: •Structural integrity was confirmed under restricted circumstances. (10) Heavy liquid metal coolant technology (Oxygen control system): •Tested and assured only for small-scale corrosion test loop in a limited condition. (11) Refueling •Very small-sized tests or preliminary experiment (12) Mechanical design (Bottom Mounted Control Rod Drives, Mechanical systems that prevent the fuel, blanket, and shutdown assemblies from floating up out of the core.) •Detail investigation is necessary	*See concept evaluation of Group C and Group D* <u>SVBR</u> (1) Passive shutdown: •Development of SASS. (3) Passive Decay heat removal: •Development of RVACS (Water Tank Type). (9) Material against Corrosion Problem: •Structural integrity over 600 C (10) Heavy liquid metal coolant technology: •Development of oxygen control system, filter, and oxygen activity sensor.	(1) Passive shutdown •SASS: Demonstration for practical use. •Inherent safety feature: Demonstration in a practical scale (2) Re-critically free: •Demonstration for practical use. (3) Passive Decay heat removal: •Demonstration of the capability of natural circulation cooling. (4) Structural design: •Development of structural design code applicable to HLMC reactor. •Higher Operating Temperatures (5) Seismic isolation: •Demonstration of three-dimensional system. (6) Innovative concept of component (Direct Contact Steam Generator): •Followings must be confirmed a. Liquid metal embrittlement-induced failure of the turbine blades and turbine casing. b. Pb or Pb-Bi erosion of the turbine surfaces. c. Contamination of the turbine plant by radioactive Polonium (7) Examination of thermo-hydraulics: •Examination for proposed systems (confirmation of degradation of thermo-hydraulic properties (hydrodynamics and heat transfer) as a function of the operating conditions. •The contact heat transfer resistance of oxide coatings function to the structural material and the operating condition (temperature, velocity, and oxygen concentration in the coolant). (8) ISI&R: •Development of ISI&R technologies in HLMC. •Countermeasure of the Alpha-active Po-210 (9) Material against Corrosion Problem: •Examination of corrosion resistance of existing materials. •Development of protection techniques and a new material. •Corrosion rate mapping as a function of temperature, coolant velocity, oxygen content and other parameters for a spectrum of possible reactor and cladding materials (both ferritic and stainless steels). •Confirmation of degradation of mechanical properties of structural materials as a function of the operating conditions. •Radiation damage Examination of candidate materials that have sufficient LBE corrosion resistance. (10) Heavy liquid metal coolant technology: •Development of oxygen control system, filter, and oxygen activity sensor. (11) Refueling •Development of Refueling technologies in HLMC. (Refueling system, removing HLMC, etc) •Countermeasure of the Alpha-active Po-210 (12) Mechanical design •Development of Bottom Mounted Control Rod Drives •Development of Mechanical systems that prevent the fuel, blanket, and shutdown assemblies from floating up out of the core.

E.3 FUEL CYCLE TECHNOLOGY

Two main types of processes can be applied to the separation of long-lived radionuclides: Aqueous and pyrochemical ("dry") processes. The industrial aqueous technique is used in the PUREX process to separate U, Pu and eventually Np (in a modified PUREX process) from dissolved spent fuel. For the extraction of minor actinides the process has to be extended, i.e., additional extraction steps follow the standard process. Extensive research is being carried out at present in this field mainly to extract americium and curium, including the separation of minor actinides from the lanthanides, which are generally co-extracted due to very similar chemical properties. An alternative to aqueous processes are pyrochemical processes in which refining is carried out in molten salt media, based on electrorefining or on distribution between non-miscible molten salt-metal phases.

The oxide fuels used world-wide in thermal reactor systems for energy production are more easily recycled by aqueous techniques; therefore these systems, using primarily the Purex process, are more fully developed and implemented commercially. Pyroprocess systems have largely been associated with fast reactors and metallic fuels and their development has therefore only reached the pilot-scale stage and the feasibility of Minor Actinide (MA) separation still needs to be demonstrated.

The major advantages of pyrochemical methods to recycle advanced fuels, in comparison to aqueous techniques, are a higher degree of compactness of equipment and the possibility to form an integrated system between reactors and fuel cycle facilities, thus reducing considerably transport of nuclear materials. Especially for advanced oxide fuels (mixed transuranium, inert matrix or composite) and metal fuels, but also nitride fuels, pyrochemistry is to be preferred. Compared with aqueous methods, dry processing of fuels results in less pure and thus likely more proliferation resistant fractions of Pu, Np or Am. In addition, the radiation stability of the salt in the pyrochemical process compared to the organic solvent in the aqueous process offers an important advantage when dealing with highly active spent MA fuel. Shorter cooling times reduce storage costs. Due to the absence of water (neutron moderator) in the process the criticality hazard is lower; i.e., the inherent safety is increased.

E.3.1 Advanced Aqueous Recycle

The PUREX process is the most important aqueous recycle technique to separate U and Pu from spent fuel containing natural, slightly or highly enriched uranium. It is employed in the spent fuel recycle industry and is taken as reference process in the LWR/UOX and LWR/MOX contexts. It applies tri-n-butyl phosphate (TBP) in a kerosene-type aliphatic hydrocarbon diluent as the extractant and utilises the highly selective extractability of hexavalent U and tetravalent Pu nitrates to separate them from the bulk of fission products (FP) in a nitric acid (HNO3) medium. Nowadays, alternatives such as crystallization of fuel solution, process without plutonium purification, etc. have been studied to simplify the PUREX process.

The present state of the art in recycling allows the selective recovery of U and Pu with efficiencies of about 99.9%, according to recent PUREX plants feedback. The minor actinides (Np, Am, Cm) and the fission products remain in the liquid HLW and are incorporated in the vitrified high-level waste.

Three different options can be distinguished in the PUREX process according to which the partitioning of TRUs (Np, Pu, Am, and Cm) can also be achieved:

<u>Standard PUREX process</u>:

U and Pu are separated with an industrial yield close to 99.9 %. MAs and FPs are conditioned in a glass matrix for interim storage and final disposal. Am and Cm are not separated because their valences (III) are not extracted by TBP. They remain in the raffinate (HLLW) of first cycle together with the bulk of fission products, except the gaseous products like Kr-85 and I-129, which are separated in the off-gas of the spent fuel dissolution. Np is partially extracted by TBP, following the U stream and it is separated from the latter in the second U purification cycle. This Np stream is added to the HLLW.

PUREX process adapted for Np recovery:

It is based on the standard PUREX process with the improved separation of Np. Neptunium is present in dissolution solutions feeding to the first decontamination cycle as a mixture of Np (IV), Np (V) and Np (VI). If Np should be co-extracted adapted Np recovery a complete oxidation to the oxidation state VI is required. The Np is then extracted together with U and Pu in the first decontamination cycle and follows the U stream after the Pu partitioning. Finally it is recovered through a reducing scrub in the second U cycle. After separation, the Np nitrate may be purified by solvent extraction with TBP and finally transformed to oxide by calcination of the oxalate.

Extended PUREX process for MA recovery:

This process includes the separation of minor actinides (Am and Cm) and some mainly long-lived fission products. Extensive R&D is carried out to develop special extractants required for the separation of minor actinides (Am and Cm) from HLLW. Three alternative approaches are proposed:

1. The first is based on co-extraction of minor actinides (III) and lanthanides (III). The developed or under-development processes are:

    - TALKSPEAK (USA) and DIDPA (Japan), which use acidic organophosphorus as extractants. Also the DIDPA process could be used to recover Np from HLLW
    - The TRUEX process (USA) and SETFICS (Japan) are based on the use of CMPO (n-octyl-phenyl-di-isobutyl-carbomoylmethyl-phosphine-oxide).

2. The TRPO process (China) uses a trialkyl phosphine oxide. This process has also been developed at the ITU (Karlsruhe).

The DIAMEX (diamide extraction) process using malonamides as extractants has been developed at CEA (France). The second alternative is based on the separation of minor actinides in a single operation, leaving all the lanthanides in the HLLW. High selectivity extractants such as TPTZ (UK-France) (tripyridyltriazine) [85Mus] or CYANEX 301 (China) (trimethyl-pentyl dithiophosphinic acid) are used sometimes in synergistic combination with TBP. New extractants and methods are under research and development, i.e., within the 5th Framework Program of the EC two projects PARTNEW and CALIXPART. The main extractants to be developed and tested (with some very significant experimental results today already obtained with genuine fuels) are diamides (DIAMEX), nitrogen polydendate ligands, dithiophosphinic acids, and acidic S, Se, or Te bearing ligands. The last three groups of compounds are used in the SANEX (selective actinide extraction) process to separate selectively the trivalent actinides (Am and Cm) from HLLW. The objective of the CALIXPART is synthesis of matrices able to extract and separate in a single step actinides from lanthanides. Several ligand chains, including nitrogen or sulphur atoms properly organised on matrices such as calixarenes and others, will be synthesised and tested for the oxidation of minor actinides and their selective extraction. This is based on the fact that in contrast to lanthanides, Am can be oxidised in nitric acid solution. However, the oxidation states IV and VI are unstable and must be stabilised with a strong oxidation reagent. The

SESAME process, developed by CEA, uses an electrochemical method to oxidise the Am and then to separate it selectively from curium, lanthanides, or other FPs.

An overview of recovery yields by the above-mentioned PUREX processes is shown in Table E-13

Table E-13. Recovery yields for PUREX Aqueous Recycling.

Elements		Standard (%)	Improved (%)	Extended (%)
U*		99.9	99.9	99.9
Pu*		99.9	99.9	99.9
Minor Actinides	Np		95–99.9	99.9
	Ammm			99.9
	Cm			99.9

* According to industrial process plant operators.
** Depending on the process parameters.

The JNC (Japan) has been studying the "advanced aqueous process," which combines the "crystallization process" with the "simplified solvent extraction process" and adds an MA collection function. The main features compared with the conventional PUREX are summarized as follows:

- The purification process of U and Pu in the conventional PUREX is eliminated, resulting in co-extraction of U/Pu/Np, and the simplification of the system. The compact-size centrifugal type equipment is adopted for the extraction process to make the facility smaller.
- The crystallization method is also adopted for recovery of excess U before extraction of U/Pu/Np, which reduces the amount of liquid treated in the extraction process. The main stream is the salt-free process, which means reduction of intermediate-level waste.
- An additional process is also considered for the recovery of Am and Cm. The combination of the SETFICS process (developed by JNC) and the TRUEX process is applied for this stage.
- According to the evaluation of the mass balance, the recovery ratio of U/TRU has been estimated to be 99.7%, and the decontamination factor of the product is higher than 10^2.

In addition, in order to improve the advanced aqueous process further, the ion exchange method, the amine extraction method, and the supercritical fluid extraction method are also investigated as an alternative or supplementary aqueous technology.

Other process developed by JAERI (Japan) aim at a separation of the elements contained in HLLW into four groups (Four-Group Separation Process):

- All TRU elements including Np (V) are extracted with DIDPA after to reduce the nitric acid concentration to 0.5 M.
- The solubilised fraction of Tc and platinum group metals are separated by precipitation through denitration.
- Sr and Cs are separated by adsorption with inorganic ion exchangers.
- After the separation of TRU by DIDPA, they are back-extracted with different reagents. Am, Cm and lanthanides are extracted again with DIDPA. Then, Am and Cm are back-extracted, leaving lanthanides in DIDPA solvent.

Some long-lived fission products such as ^{129}I, ^{99}Tc could also be separated within an improved PUREX process. Special off-line separation techniques are therefore necessary to separate Am and Cm and possibly Cs and Sr from liquid HLW.

Figure E-4 gives an overview of an advanced recycling of spent fuel based on aqueous processes. The advanced aqueous process proposed by JNC (Japan) and the alternative or supplementary aqueous processes are shown in FigureE-5 and FigureE-6, respectively.

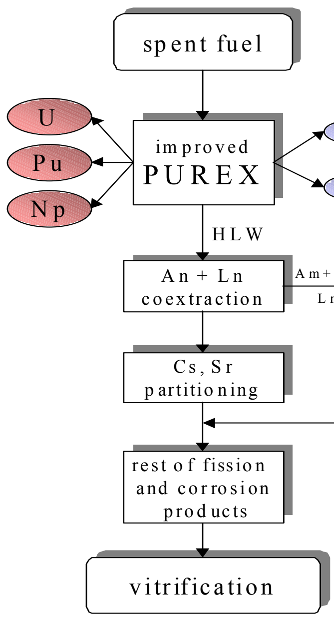


Figure E-4. Advanced reprocessing of spent fuel, including MA recovery and separation of some long-lived radionuclides.

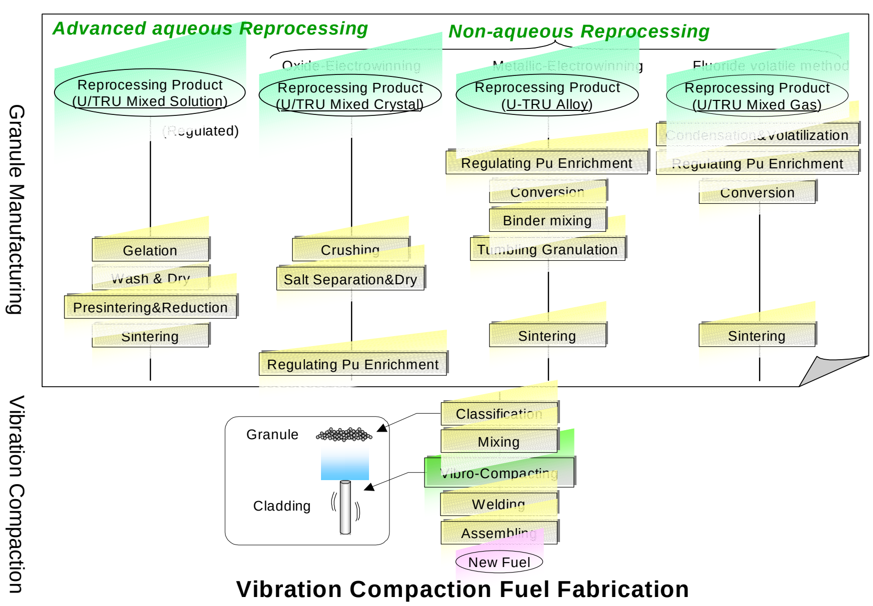
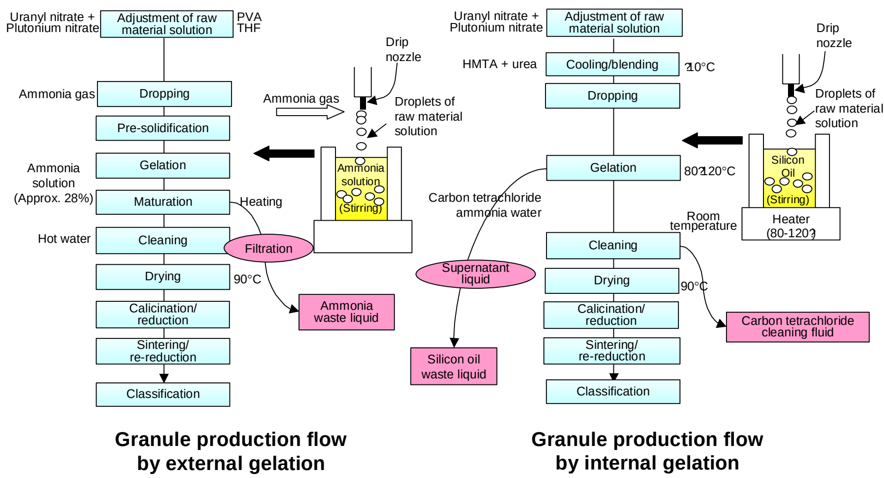


Figure E-5. Alternative or supplementary aqueous process.



Particle Fabrication that Matches with Aqueous Reprocessing

Figure E-6. Alternative or supplementary aqueous process.

With respect to aqueous recycling, dedicated research facilities are available in France (CEA, Marcoule) and the United Kingdom (BNFL, Sellafield), where extensive studies are being performed on spent fuel material. Facilities for smaller-scale hot test are available in other countries and the JRC-ITU (Karlsruhe, Germany). In addition, European Universities are strongly involved in this research through the Framework Programmes, especially in the fields of synthesis of new extractant and molecular modelling.

E.3.2 Fabrication Technology Supporting Advanced Aqueous Recycling

Fabrication technologies must support the proliferation-resistant, economical fuel cycle system, which by definition means fabrication technologies must be available for low-decontamination factor processes. Also from the viewpoint of minor actinide (MA) recycle to reduce the MA amount in the high active waste, it is important to develop fabrication technology for MA-bearing fuel.

The MA-bearing and the low-decontaminated material resulting from advanced recycling has too high a radiation activity to fabricate fuel by contact operation in a glove-box facility. Therefore, the nuclear material must be handled in a hot-cell facility to produce fuel pellets, particles or slugs, fuel pins, and finally fuel assemblies by completely automated equipment with remote maintainability.

The candidates for the fuel fabrication technology are described in this section, as follows.

1. Simplified Pelletizing Method for FR-MOX Fuel Fabrication

(1) Objective and progress of the technology development [1]

The Japan Nuclear Cycle Development Institute (JNC) has been conducting research and development in MOX fuel fabrication for fast reactors (FRs). First, the Plutonium Fuel Production Facility (PFPF) with a capability of 5 ton-MOX/year was constructed in 1987 to demonstrate the mass production technology of MOX fuel, employing remote and automated operation technologies. Up to the present, the PFPF has fabricated more than 400 MOX fuel assemblies in total for the FR MONJU and JOYO, and the cumulative amount of MOX fuel has reached about 12 tons of MOX. It is considered that the MOX fuel pellet fabrication technology in glove boxes with highly decontaminated materials from the current aqueous fuel cycle plant has already been established through these fabrication experiences.

However, it is essential to the realization of commercially accepted FR fuel cycle, with costs comparable to that of LWRs, to simplify the current MOX fuel fabrication process. Therefore, the JNC started to find fundamental process conditions to realize the advanced fuel fabrication process called “Short Process” or “Simplified Pelletizing Method,” by laboratory-scale experiment in 1999.

Further, JNC started the feasibility study on commercialized fast reactor cycle systems in cooperation with Japanese electric utilities, Central Research Institute of Electric Power Industry (CRIEPI) and Japan Atomic Energy Research Institute (JAERI) in July 1999. In the feasibility study, by using the excellent features of FRs, which can burn minor actinides and low decontaminated fuel, it is expected in the future to reduce the fuel cycle cost and to increase the proliferation-resistance by means of adoption of a low decontamination factor in processing. As a result, the nuclear materials are required to be handled in a hot-cell facility to produce fuel particles or pellets, fuel pins and assemblies by completely automated equipment with remote maintainability. The Simplified Pelletizing Method mentioned above is basically suited for the fabrication in cells because it would enable easier remote-maintenance for process equipment due to a drastic reduction in the number of process steps. Therefore, JNC adopted the Simplified Pelletizing Method as one of the promising candidates for MOX fuel fabrication technologies adequate for the future FR recycle system, and started the conceptual design study with the Simplified Pelletizing Method for a commercial-scale fabrication plant.

(2) Outline of the process

The process flow diagram of the Simplified Pelletizing Method compared with the conventional process is shown in figure E-7. The following pin loading and bundle assembly process is the same as the conventional one without the rework process.

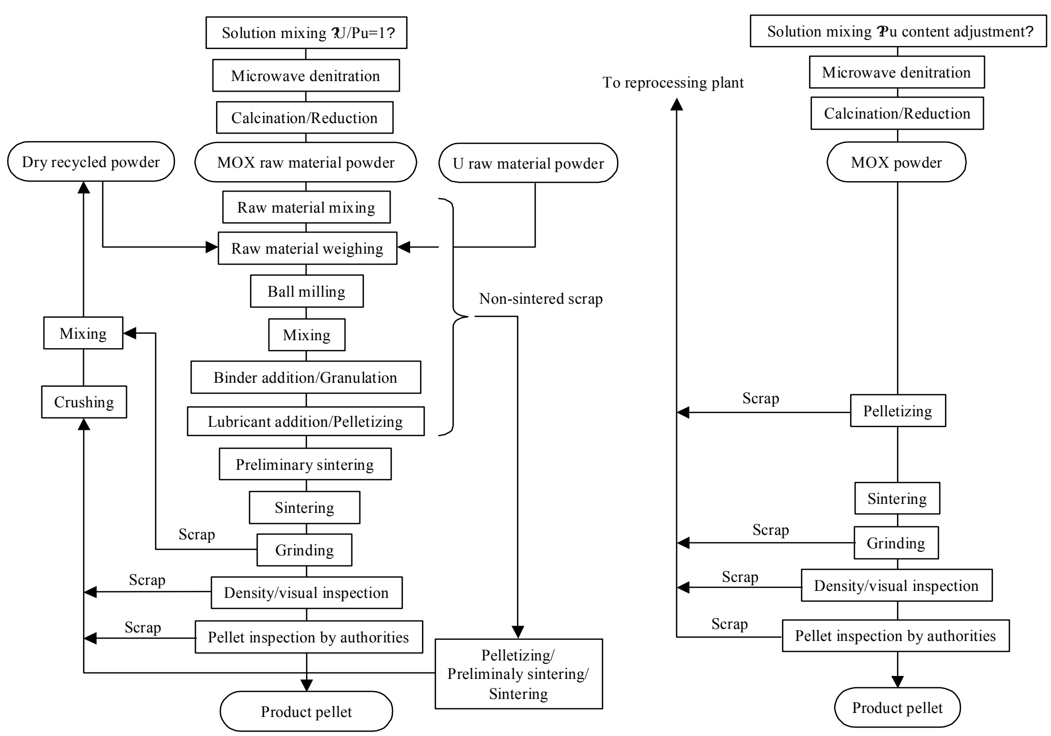


Figure E-7. Comparison of the simplified palletizing method with a conventional one.

In the advanced process, the plutonium contents satisfying the fuel specification is adjusted in a mixing step of plutonium nitrate and uranium nitrate in a recycle plant. And then the mixed solution is converted into MOX powder at a conversion facility by using the microwave direct denitration method. The property of MOX powder is prepared by the calcinations and reduction to have certain flowability that can be palletized into green pellets without granulation. Nuclear materials generated from rejected pellets and pins are transferred to the wet-recovery process in the plant.

After the realization of the advanced process, it is possible to skip a number of process steps and also achieve the same quality level in the products, compared with the conventional process.

(3) Technical Features

By introducing the Simplified Pelletizing Method, the following benefits are expected.

- The fuel fabrication cost and the production of radioactive waste will be drastically reduced.
- By eliminating the powder blending and granulation steps from the conventional MOX pellet process, the mass of MOX powder falling and scattering in the process will be reduced. It results in increasing lifetime of process equipment because the dispersed MOX powder promotes the deterioration of cables, electronic equipment, etc. Further, it decreases personnel radiation exposure during maintenance operations.

(4) Current status of technical development [2]

The following fundamental technologies are being investigated and developed to assure the availability of the Simplified Pelletizing Method.

- Adjustment technique of plutonium contents during the mixing stage of uranium and plutonium nitrate-solutions in a recycling plant,
- Powder flowability enhancement technique, for example by controlling the temperature during calcinations/reduction,
- Pressing equipment with dies-wall lubrication,
- Pneumatic powder transport systems including the accountability system for nuclear materials.

(5) Technology to be developed

For the establishment of the Simplified Pelletizing Method, it is necessary to develop the above-mentioned technologies.

In addition, for applying the Simplified Pelletizing Method to the MOX pellet fabrication in cells, it is necessary to develop the following technologies.

- Remote-maintenance,
- Handling of low decontaminated TRU fuel including heat removal measures.
- A turntable type denitration/calcinations/reduction system:

  In order to eliminate the powder blending and granulation step from conventional MOX pellet processes, the operation on MOX powder preparation process should be more reliable than the conventional one because its reliability is affected the rate of the system operation. Therefore the compact equipment concept of the conversion process includes microwave denitration calcinations. The equipment's function is to convert the materials from a nitrate solution to MOX powder in the same container, and then transfer to the next process step on the rotation table. A devise has been designed based on a design philosophy of minimum transfer operations, which will minimize dust generation, and of simple mechanisms to have remote-maintainability.

(6) References

[1] K. Asakura et al., "Current Development of Fuel Fabrication Technologies at the Plutonium Fuel Production Facility, PFPF," Int. Symp. on MOX Fuel Cycle Tech. for Medium and Long Term Development, IAEA-SM-358/4 (1999).

[2] "Feasibilities on Commercialized Fast Breeder Reactor Cycle System (Phase I) Interim Report," JNC TY 1400 2000-004 (Aug 2000), in Japanese.

2. Vibration Compaction Fuel Fabrication Technology

(1) History of development

The first trial of this technology was made at Oak Ridge National Laboratory (ORNL) in the United States in the late 1950s. At that time, the sol-gel process was developed for thorium oxide. At an early stage of development, particles obtained were irregular-shaped granules (shards), resulting in large stresses to the cladding during compaction, which would induce the degradation of the cladding. Because of this, vibration-compaction fuel pin fabrication was thought to be difficult.

In the early 1960s, it was realized that well-shaped, highly dense, spherical particles could be produced by applying the gel controlling technique, which had been developed in the catalysis industries. Vibro-packing of spherical particles with relatively low vibration energy was tried.

In Europe, activity of fundamental research and development of the sol-gel method also began in the 1960s.

In the 1970s, a gelation process was developed, by which it became possible to produce spherical UO_2 particles with the diameter larger than 600 micrometers, which had been difficult by the sol-gel process. Since then, many pilot plants were constructed and operated, and also small-scale irradiation experiments were performed in Italy, the United Kingdom, Germany, and the United States, etc. [1]

In the United Kingdom, irradiation experience reached about 700 pins in the Dounreay Fast Reactor (DFR) and about 3000 pins in the Prototype Fast Reactor (PFR). [2] In the United States, several pins were irradiated with burnup up to 12 at.% in EBR-II. [3] In Switzerland, carbide fuel was irradiated up to about 10 at.% of burn-up. [4]

Late in the 1980s, as development of the fast reactor slowed, development activities of vibration compaction method were also decreased.

In Japan, about the end of 1962, development of the sol-gel method started in the predecessor company of JNC. Development activity, however, did not reach full scale. Fundamental research was also made for ThO_2 fuel at the Japan Atomic Energy Research Institute (JAERI).

In Russia, development of vibration compaction fuel fabrication combined with the oxide electrowinning method (nonaqueous recycle) started in the 1970s at the Research Institute of Atomic Reactors (RIAR). In 1977, a pilot plant was constructed. Up to now, 426, 10 and 2 MOX fuel assemblies were irradiated in BOR-60, BN-600, and BN-350, respectively (about 18000 fuel pins in total). Maximum burnup in BOR-60 was about 32 at % for test pins. [5] RIAR also has experience to produce granules by vapor oxidation of fluorides. [6]

(2) Concept of the technology

The gelation process for an aqueous recycling system is categorized as either external gelation or internal gelation. In both cases, the process sequence is similar. Namely, a specific organic agent is added first to an aqueous solution containing heavy metals. Then, the solution is dropped as a droplet into another solution or bath. In the course of this process, droplets become a spherical gel.

In the Russian process, applied to the oxide-electrowinning method, U and Pu dissolved in chloride salt solutions are precipitated at an electrode as oxides, and such oxide granules are crushed and compacted into a fuel rod.

The vibration technologies are categorized into the following three methods, i.e., electrical vibration, pneumatic vibration, and electromagnetic vibration. The electromagnetic vibration method is applied both to internal gelation in the Paul Scherer Institute (PSI) in Switzerland, and to vi-pac fuel in Russia.

In the case of sphere-compacted fuel, packing density is determined geometrically by the diametrical ratio of the spherical particles, and the number of different diameters of particles. On the other hand, in the case of irregular-shaped granules, edges of granules are ground by the vibration energy, which makes packing density higher.

The main development item of the spherical packing (sphere-pac) technique is the improvement of the smear density. For irregular-shaped vibro-compacted fuel, it is important to find the vibration condition that will minimize scarring on the inner surface of the cladding.

(3) Characteristics of the technology

It is said that remote operation would be easily attained for sphere-pac fuel fabrication because the process does not include powder-handling steps, and as a result there will be no dispersion of fine powders.

Since the solution material of external gelation is thermally stable, the process of external gelation is considered to be simple and efficient. On the other hand, the amount of waste solution produced is relatively large, and the sphere is likely to form a shell structure.

Granules produced by the internal gelation method have a high degree of sphere-like shape and homogeneity. On the other hand, the solution material is thermally unstable, so an appropriate cooling system for the equipment is required. Further, the processing of used silicon-oil bath will be needed.

Vibropac fuel fabrication by pyroelectrochemical processing is characterized by a small number of steps, due to its combination with processing, so that it would be suitable to a scale-up of the facility. On the other hand, there is possibility that powder may be dispersed when granules are crushed. Use of chlorine when materials are dissolved is taken as one of the inferior features of this method. In this process, oxygen getter made of metallic uranium powders is used to control the oxygen potential in fuel rods.

(4) Present status of technological development

Technological development is being done in Switzerland, Russia, and Japan, etc., at the moment.

At PSI in Switzerland, researches on the sphere-pac fuel fabrication method, and on irradiation of fuel rods, are performed based on the internal gelation method.

In Japan, JNC started research on gelation in 1990, and, at the moment, is proceeding to the collaborative study on fuel fabrication with PSI.

At RIAR in Russia, irregular-shaped vipac fuel fabrication by the pyroelectrochemical method and its irradiation in BOR-60 is being continued.

(5) Items to be developed

With regard to the gelation method, establishment of the optimum condition of gelation and treatment of the waste solution are to be solved. The way of solving such items and more efficient granulation method are to be developed.

In the case of low-decontaminated and TRU fuel, it will be necessary to show the applicability of gelation even to the multicomponent systems.

Regarding the oxide electrowinning method, the Pu-enrichment distribution within much of the fabrication process should be under quality control.

The common items for aqueous and nonaqueous recycling to be developed are optimization of vibration conditions in order to attain high density, nondestructive inspection method for low decontaminated fuel, and irradiation experiments to confirm the good irradiation performance.

(6) References

[1] R. L. Beatly et al., ORNL-5469, ORNL, 1979.

[2] "Plutonium Fuel Technology," Atomic Energy Society of Japan, 1998 (in Japanese).

[3] M. J. Lackey, J.E. Selle, ORNL-5469, ORNL, 1978.

[4] JNC TY1400 2000-004, Aug. 2000.

[5] V. B. Ivanov et al., *Proc. Int. Conf. Future Nuclear Systems – Challenge Towards Second Nuclear Era with Advanced Fuel Cycles, Oct. 5-10, 1997, Yokohama, Japan* (1997), vol. II, p.1093 (Global '97).

[6] V. B. Ivanov et al., *Proc. Int. Conf. Future Nuclear Systems – Challenge Towards Second Nuclear Era with Advanced Fuel Cycles, Oct. 5-10, 1997, Yokohama, Japan* (1997), vol. II, p.1099 (Global '97).

E.3.3 The Pyroprocess Fuel Cycle

Fuel recycle technologies are grouped into two categories, aqueous and nonaqueous or "dry" processes. The pyroprocess options fall into the nonaqueous group. A number of dry processes have been investigated or are now being investigated globally for nuclear power applications.

In the concept submittals to TWG 3, 21 concepts of the 33 submitted referred to dry processes as either the reference or backup fuel cycle technology. In 18 of these 21 cases, the specific dry process referred to was Argonne National Laboratory's pyroprocess, and, for that reason, it will be taken as the basis for discussion in this section, with only passing mention of alternatives. The pyroprocess involves high-temperature operations and generally uses molten salts and liquid metals.

Pyroprocess systems have largely been associated with fast reactors, since the recovered fissile material is incompletely decontaminated and not easily recycled into thermal reactors. For this reason, aqueous processes, primarily the PUREX process, were adapted for use in thermal reactors. Because of this, but mostly because its development originated in military activity, PUREX at least is far more fully developed as a recycling technique than any of the nonaqueous alternatives.

In contrast to the pure extractions of conventional PUREX, the inability of the pyroprocess to recover pure fissile material is now considered an advantage with respect to proliferation resistance and is one of the prime reasons for its resurgence over the last two decades. The beginning of this resurgence occurred during the 1980s with development of the Integral Fast Reactor (IFR) concept, which employed the pyroprocess fuel recycle [1]. Pyroprocess technology, which has also been referred to variously as the pyrochemical process, the pyrometallurgical process, and electrometallurgical treatment, has been developed internationally for the recycle of metal, oxide, and nitride fuels, as well as treatment systems for the disposal of various spent nuclear fuel types.

In addition to the nonproliferation advantages, pyroprocess systems have other advantages over aqueous systems. The solvents in these systems are not susceptible to radiation damage and degradation. Compared to the conventional aqueous process, there are very few process steps, and the equipment and the facility are much more compact. Also, unlike aqueous processes, the pyroprocess is a *batch* process, rather than *continuous*. However, relatively large batch sizes are possible because the absence of moderator anywhere in the process cell allows large batches from a criticality perspective. The recycled fuel needs to be remotely fabricated because of the low decontamination factors [2], which is both a proliferation advantage and a throughput disadvantage. The main disadvantage, though, is that development has only reached the pilot-scale stage.

In the U.S. context, in the mid-1980s and continuing today, there is advantage to processes that can be economic at small scale, i.e., that do not depend on large economies of scale for economic competitiveness. This is a big advantage in avoiding cost penalties for fuel cycle service of the first few plants. In the United States, it may be true that only with such an approach can initial startup deployment be contemplated. The pyroprocess has that potential.

Pyroprocess Technology

United States

Many of the pyroprocessing systems presently proposed for development are spin-offs of industrial metal processes. This general fuel cycle is depicted in Figure E-8.

The fuel is recycled using an electrochemical process that employs molten salts and liquid metals. The molten salt medium for electrorefining is a solution of LiCl-KCl eutectic and dissolved actinide chlorides, such as UCl_3. The operating temperature is 500°C. With this system, chopped spent fuel is loaded into the electrorefiner in baskets. The fuel is electrochemically dissolved into the system in an operation in which the baskets are the anodes and another electrode in the salt phase is the cathode. Uranium with little TRU material can be collected on steel electrodes (solid cathodes), and TRU materials can be co-deposited with uranium in liquid-cadmium cathodes. A liquid-cadmium cathode is a ceramic crucible containing molten cadmium that can be lowered into the salt phase. The cadmium in the crucible is at cathodic potential. Because of the chemical activities of the TRU elements in cadmium, they can be easily deposited with uranium in liquid-cadmium cathodes but not on solid cathodes. The cathode products from electrorefining operations are further processed to distill adhering salt and cadmium and to consolidate the recovered actinides. The recovered actinides are remotely fabricated into new fuel for recycle.

The alkali, alkaline earth, rare earth, and halide fission products are primarily in the salt phase. The elements that distribute into the salt phase are eventually disposed in a ceramic high-level waste. More than 90% of the noble metal fission products and fuel alloying material (zirconium) are retained in the chopped cladding segments in the anode baskets. The cladding hull segments and the retained fission products are eventually stabilized into a metal high-level waste.

Adaptations of this technology exist for the treatment of both oxide and nitride fuels. The flowsheet for the treatment of nitride fuels is similar to that of metal fuel. In this system, the nitride fuels are also fed directly into the electrorefiner. The actinides are dissolved from the fuel cladding and collected electrochemically in liquid cadmium or bismuth cathodes. Nitrogen is evolved in the process. It is collected and recycled back into the liquid cathodes so that actinide nitrides are formed, a potentially difficult step. After distillation of the cadmium, the recovered nitrides are sized and then fabricated into new fuel using vibro-packing. This process is being developed in Japan.

A system for oxide fuels was developed so that light water reactor fuels could be converted to metals and the recovered fissile material used in metal fuel fast reactors. A head-end oxide reduction step is needed for this conversion. The reduction step is typically performed chemically in a similar salt as used for electrorefining. In an example process, the oxide fuel is reduced to metal by reaction with lithium dissolved in LiCl at 650°C. The recovered metal would then be subjected to electrorefining as described earlier. The Li_2O would be converted back to lithium metal by electrowinning. Pyroprocessing options are also being explored to treat oxide fuels in a manner in which the recovered material is still in an oxide form [3]. Extensive work on this cycle has been performed at the Russia Institute of Atomic Reactors [4,5]. The vibro-packing technology for fuel fabrication was included as part of this development.

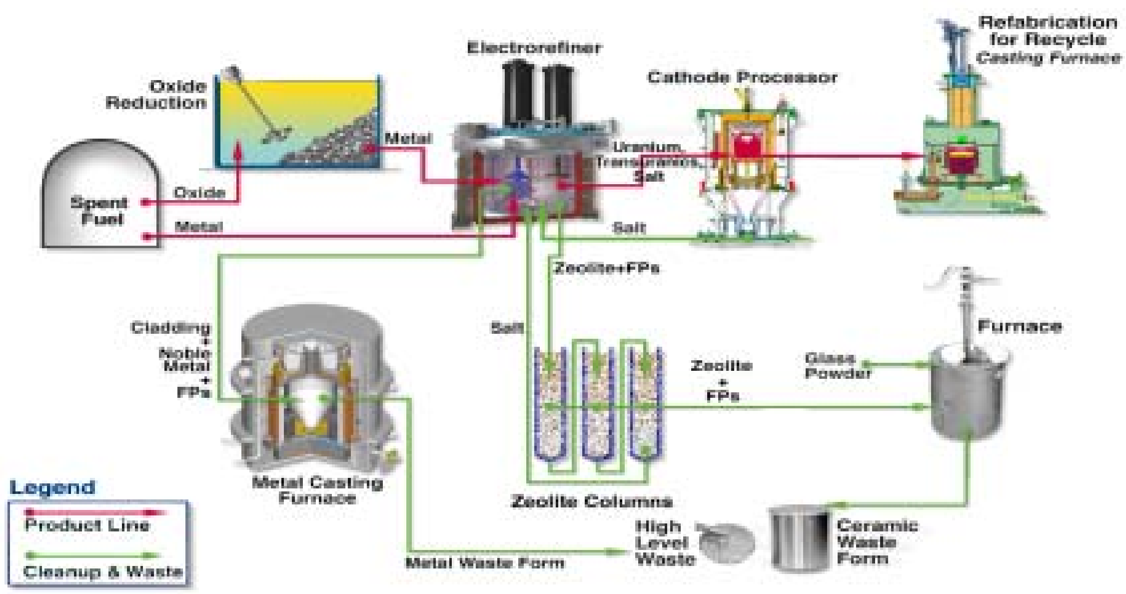


Figure E-8. Metal and oxide fuel pyroprocess flowsheet.

JAPAN

The JNC started the feasibility study on commercialized fast reactor cycle systems in cooperation with Japanese electric utilities, CRIEPI and JAERI in July 1999. In this study, “oxide electrowinning,” “metal electrorefining,” and “fluoride volatility” have been evaluated as pyrochemical methods for three types of fuels: oxide, nitride, and metal fuels. Improvements were made to the original process, as discussed below, and technical feasibility of the improved process flow was reviewed in more detail.

(1) Oxide electrowinning (Figure E-9)

The original method was developed by RIAR. Improved points for oxide fuel recycling are as follows:

- The salt electrolyte composition of NaCl-2CsCl instead of NaCl-KCl in order to lower the operation temperature

- Improvement of processing speed and reduction of chlorine consumption by adopting simultaneous electrolysis
- Addition of a process to separate platinum-group (noble metal) FPs that affect reactor core performance negatively
- Co-deposition of uranium and plutonium-mixed oxides instead of precipitation of pure plutonium dioxide
- MA recovery by drawing electrolysis (extra electrowinning stage).

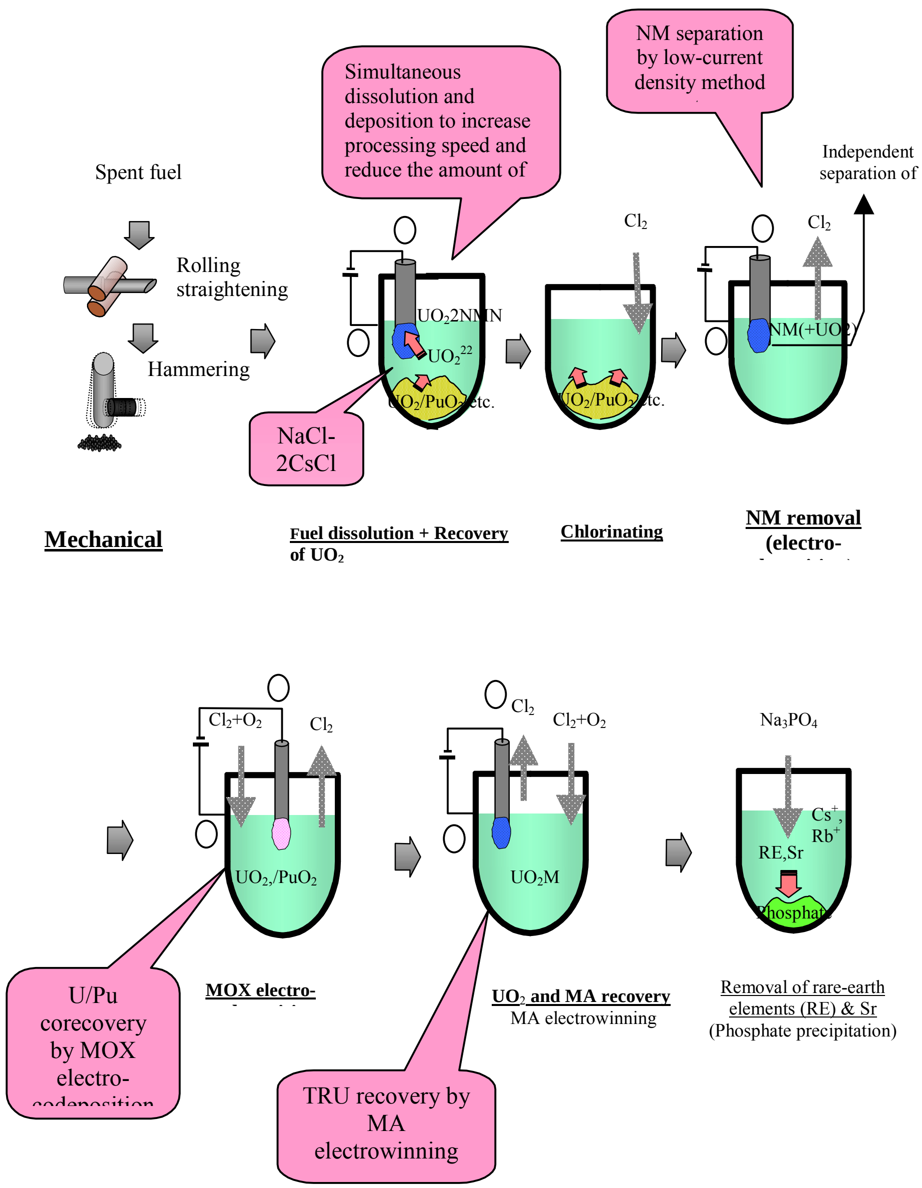


Figure E-9. Oxide electrowinning (oxide fuels).

(2) Metal electrorefining (Figure E-10 and FigureE-11)

The original method was developed by ANL (recovery of uranium and plutonium with MA). Improved points for oxide fuel reprocessing are as follows:

- FP separation in the decladding process by adopting thermal decladding method
- Reduction of salt waste and lower operating temperature by adopting of dissolution process by chlorine gas(however, additional platinum-group FP (NM) separation process is required.), or
- Reduction of Li salt waste by recycling of Li in case of adopting Li reduction process as an alternative technology
- MA recovery by multi-stage extraction

Improved points for metal fuel recycling are as follows:

- Reduction of salt waste (increasing lifetime of salt)
- MA recovery by multi-stage extraction

(3) Fluoride volatility

The process was improved in the following points:

- Introduction of co-recovery of uranium and plutonium oxides with low decontamination level,
- MA recovery by using fluoride molten salt and nitric acid.

On the basis of the above studies, the oxide electrowinning method (fuel type is oxide fuel) and metal electrorefining method (fuel types are oxide, nitride and metal fuels) were selected as the pyrochemical technologies to be studied .in the following Phase 2 of the feasibility study. The fluoride volatility method is placed as a technology to be reevaluated in Phase 2 by reviewing results of other studies in Japan and abroad, and to be studied the possibility of hybridisation.

At the JNC the examination of pyrochemical processes is performed on the laboratory and on the engineering scale and aims at an evaluation of the potential of this technology from an economic point of view. Comparative assessments between dry processes and aqueous processes will be done in the near future at JNC in collaboration with utilities, CRIEPI and JAERI.

A glovebox facility aiming for plutonium behaviour examination with a capacity to treat 200 g of fissile material (MOX) is installed in Tokai-mura. The Pu behaviour is studied in each step of the pyrometallurgical process, i.e., fuel reduction, electrorefining, liquid-liquid extraction, chloride distillation, metal distillation, oxidation. Emphasis is mainly put on the performance and the safety aspects of the process.

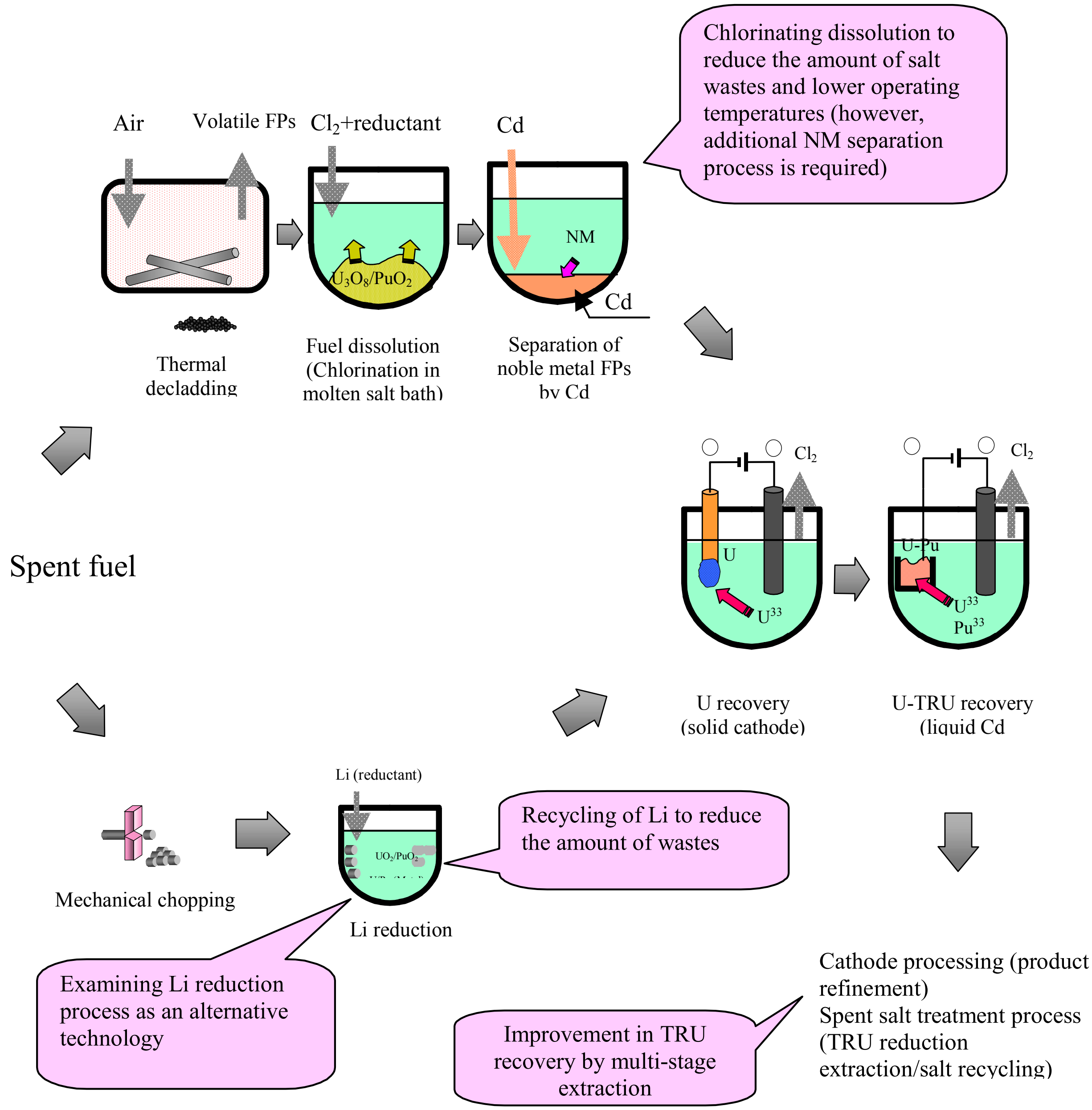


Figure E-10. Metal electrorefining (oxide fuels).

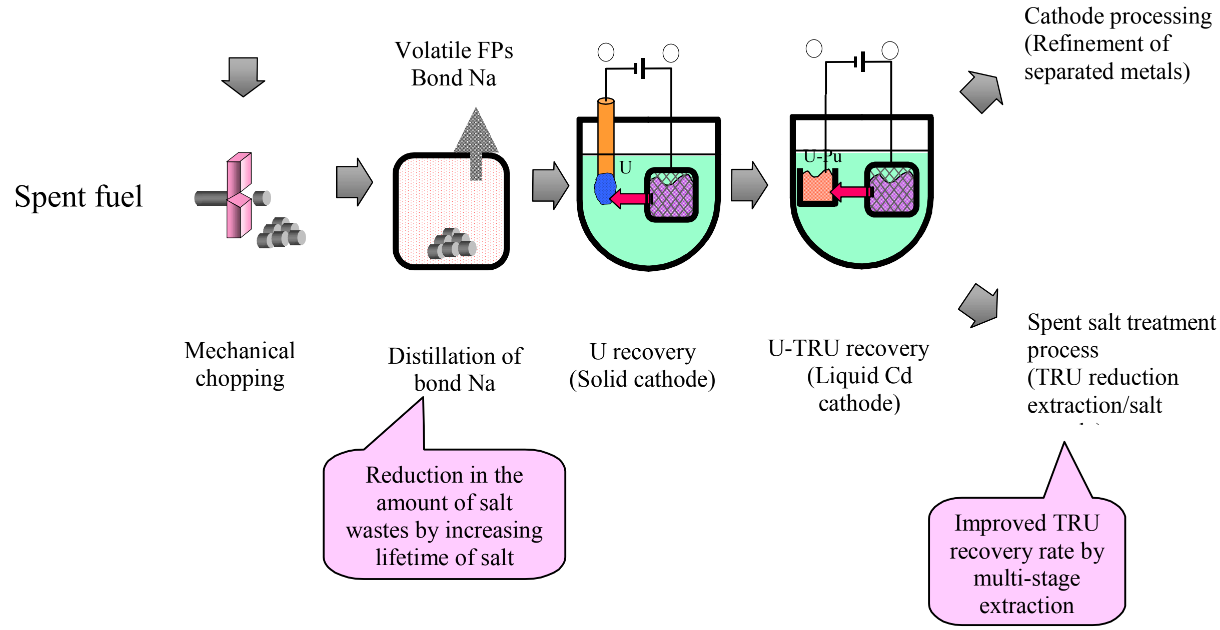


Figure E-11. Metal electrorefining (metal fuels).

Russia

In Russia, the Research Institute of Atomic Reactors (RIAR) in Dimitrovgrad has successfully demonstrated the technical feasibility of the pyroelectrochemical reprocessing of oxide fuel by direct electrorefining without reduction. Several kg of spent fuel from the BOR 60 fast breeder reactor has already been reprocessed.

Russian experience with molten salt dates from the 1950s with creation of the High-Temperature Electrochemistry Institute at Ekaterinbourg. A wide range of laboratory studies (characterization of salt structures, electrochemistry, kinetics, etc.) was carried out there in molten chloride media on uranium, thorium, and simulated fission products. Work began in 1964 on plutonium in the Electrochemical Processes Laboratory of the RIAR-Institute to specify a series of elementary operations for the fabrication and reprocessing of UO_2 and MOX fuels by the vibratory compaction method.

Research activities today (potentiometry, voltammetry, etc.) focus on the behavior of neptunium and americium for the fabrication of oxide fuel containing these elements. In the so-called DOVITA-process developed in Dimitrovgrad, the oxide fuel is converted into chlorides. UO_2, PuO_2, as well as (U, Pu, Np) O_2 are separated by electrolysis in a melt of NaCl-KCl at 650°C. The transuranium elements are precipitated sequentially as the oxychlorides or oxides from the NaCl-KCl melt by gassing with Cl_2/O_2 and adding Na_2CO_3. Since lanthanides and the transplutonium elements (Am, Cm) have similar behavior, a fractionated precipitation of the oxychlorides is proposed in order to obtain an Am, Cm fraction with a sufficiently low lanthanide content. From the technological point of view, however, this is a cumbersome step; therefore, it would be certainly preferable to develop an electrorefining process also for Am and Cm. The Arzamas and Kurchatov Institutes have also proposed a molten salt reactor concept for Pu and minor actinide management purposes.

Europe

A network of six European research organisations has submitted an R&D proposal on pyrochemical recycling to the 5th Framework Program for Technological Research and Development of the European Commission. The partners are CEA (Marcoule, France), together with the ENSCP (Paris), the INPG (Grenoble) and the IUSTI (Marseille), CIEMAT (Madrid, Spain) ENFA (Casaccia, Italy), ITU/JRC (Karlsruhe, Europe), together with CRIEPI (Tokyo, Japan), NRI (Rez, Czechoslovakia) and BNFL (Sellafield, UK), together with AEA-T (Harwell, UK). This program should revive European research in the area of pyrochemical recycling and contribute to establishing a long-term European expertise in this area.

Fabrication of Metal Fuel Assemblies for Recycle

Referring again to Figure E-8, metal fuel slugs are fabricated by injection casting, a process that lends itself well to remote operations. In the injection casting process, a fuel alloy is prepared by co-melting the fuel constituents in a crucible. The molten alloy is then injected into evacuated quartz molds where it is immediately quenched. The slugs are removed from the molds and cut to length. No machining of the slugs is required. Any reject slugs and the cropped ends are recycled into subsequent castings. One-piece slugs sufficient in length for the EBR-II core height (~34 cm) were cast routinely, and experimentally it was straightforward to cast slugs over 45 cm in length.

Slugs once cropped to length are loaded into cladding jackets already loaded with a small amount of metallic sodium, which acts as a thermal bond between the fuel and the cladding. The cladding is heated to allow the slugs to settle to the bottom of the jacket. An end cap is welded on, and the entire fuel pin is then baked and impacted from the bottom to eliminate bubbles. Following an eddy current test for quality of bond, the finished fuel pin is ready for fuel assembly fabrication. More than 200,000 metal fuel pins were manufactured by this method. Of these, 35,000 were manufactured remotely in a hot cell in the 1960s.

Nevertheless, development is needed to replace or accommodate the quartz molds. Even though coated with a thin yttria or zirconia wash, bits of metal adhere to the mold, which then produce a small but persistent stream of heavy metal requiring secondary treatment. Advanced mold materials, thin zirconium metal molds, and simple pyrochemical processing were all being investigated when the IFR program was stopped.

An alternative casting process, centrifugal casting, has been investigated in Japan. Argonne had first used centrifugal casting to manufacture EBR-I fuel slugs in the late 1940s. It later gave way to injection casting at EBR-II because it required more complicated equipment and because the product yield was higher with injection casting. However, the secondary stream issue, if not solved by advanced materials or some other simple approach, renews interest in centrifugal casting.

Refer to figure E-12. Ingredients of the fuel alloy are put into a crucible and melted by induction heating. Molten fuel alloy is poured into the center of a rotating mold assembly, and it is forced into the molds by centrifugal force. After cooling, the mold is disassembled, and the slugs removed and cut to length.

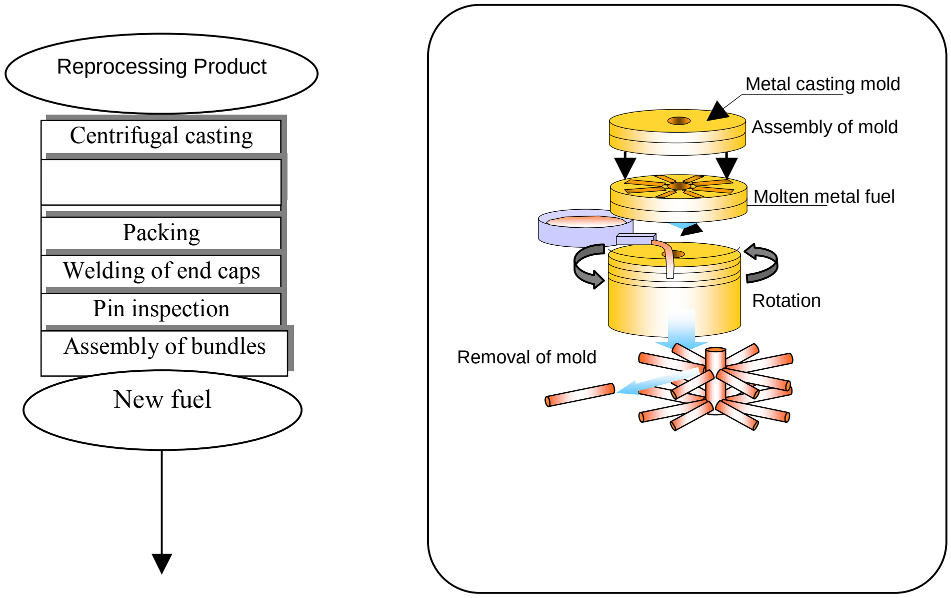


Figure E-12. Process of centrifugal casting method.

Status of the Pyroprocess Fuel Cycle

The IFR program was terminated in 1994, prior to demonstrating the technology through the recycle of spent fuel from the Experimental Breeder Reactor II (EBR-II). When this program was terminated, the pyroprocess was modified for the treatment of the EBR-II fuel for eventual disposal. The key difference between the use of the technology for fuel treatment versus fuel recycle is that the transuranics are not recovered for fuel treatment. They are instead allowed to build up in the electrorefiner salt phase and then eventually disposed of in the resulting ceramic high-level waste.

The spent fuel treatment technology was successfully demonstrated with EBR-II fuel. During this demonstration by Argonne conducted between June 1996 and August 1999, 100 EBR-II driver (400-kg highly enriched uranium) and 13 EBR-II blankets (600-kg depleted uranium) assemblies were treated [6]. This small, but representative quantity of fuel provided adequate fission products and transuranics to produce waste forms for characterization. A subcommittee of the National Research Council (NRC), established to review the progress and to evaluate the demonstration results, noted the following in their final report:

Finding: The Committee finds that ANL has met all of the criteria developed for judging the success of its electrometallurgical demonstration project.

Finding: The Committee finds no technical barriers to the use of electrometallurgical technology to process the remainder of the EBR-II fuel [7].

With the completion of the demonstration review by the NRC and a positive nonproliferation assessment [8], the Department of Energy (DOE) decided to use this technology to process the remaining EBR-II fuel (approximately 25 tonnes) and some sodium-bonded metal fuel from the Fast Flux Test Facility (FFTF) [9,10]. After completion of an environmental impact statement, these production operations started in September 2000.

The work performed to date on the treatment of nitride and oxide fuels has been on either the laboratory or engineering scale [3, 11-16]. The feasibility of the processes has been demonstrated, but large-scale tests have not been performed with irradiated spent fuel.

Technical Uncertainties

The Spent Fuel Treatment Program at Argonne National Laboratory demonstrated many parts of the pyroprocess fuel cycle, but there are still key aspects that have yet to be demonstrated on a large scale with radioactive materials. The main outstanding issue is recovery of transuranics. Large-scale equipment has been fabricated for transuranic recovery, but with the termination of the IFR program, the equipment and process was never tested beyond laboratory scale.

The remote fabrication of IFR fuel was not part of the Spent Fuel Treatment Program, but the same technology was used to fabricate cold fuel for EBR-II, and a demonstration of another pyroprocess (melt refining) for recycling EBR-II in the 1960s employed remote fabrication for 34,500 fuel elements [17].

One challenge for a pyroprocessing system is selecting the appropriate materials of construction for the high-temperature processes. Material improvements are needed in order to lessen the formation of dross streams and increase material recovery and throughput.

The quantity of waste generated that requires geological disposal from pyroprocessing appears to be comparable at present to modern commercial aqueous processes. Advancements are being pursued to further reduce the disposal volumes through zeolite ion exchange processes. This technology has not been demonstrated beyond laboratory scale.

Most radioactive work performed to date has been on the pyroprocessing cycle for metal fuel. Laboratory work has been performed on the head-end operations for oxide reduction and on the nitride fuel cycle. Demonstrations of these technologies with actual spent fuel are still needed. Additionally, for nitride fuels, demonstrating the recycle of nitrogen is critical since ^{15}N is specifically required for the fuel in order to eliminate the formation of radioactive ^{14}C.

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